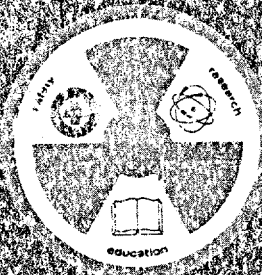


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Applied Health Physics and Safety Annual Report for 1975

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OAK RIDGE NATIONAL LABORATORY
OPERATED BY UNION CARBIDE CORPORATION FOR THE ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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HEALTH PHYSICS DIVISION

APPLIED HEALTH PHYSICS AND SAFETY ANNUAL REPORT FOR 1975

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J. A. Auxier, Director

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D. M. Davis, Associate Director

08

J. E. Turner, Associate Director

AUGUST 1976

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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FOREWORD

This report describes and summarizes the activities of the applied sections and/or groups of the Health Physics Division. Projects and activities within the research sections are described in ORNL-5046, Health Physics Division Annual Progress Report, Period Ending June 30, 1975.

Information in this report was contributed by, and/or compiled by, the following staff members of the Applied Health Physics and Safety Sections:

Radiation Monitoring

E. D. Gupton
D. R. Clark
L. C. Henley
J. R. Muir
W. W. Parkinson

Environmental Monitoring

D. G. Jacobs
W. D. Cottrell
T. W. Oakes

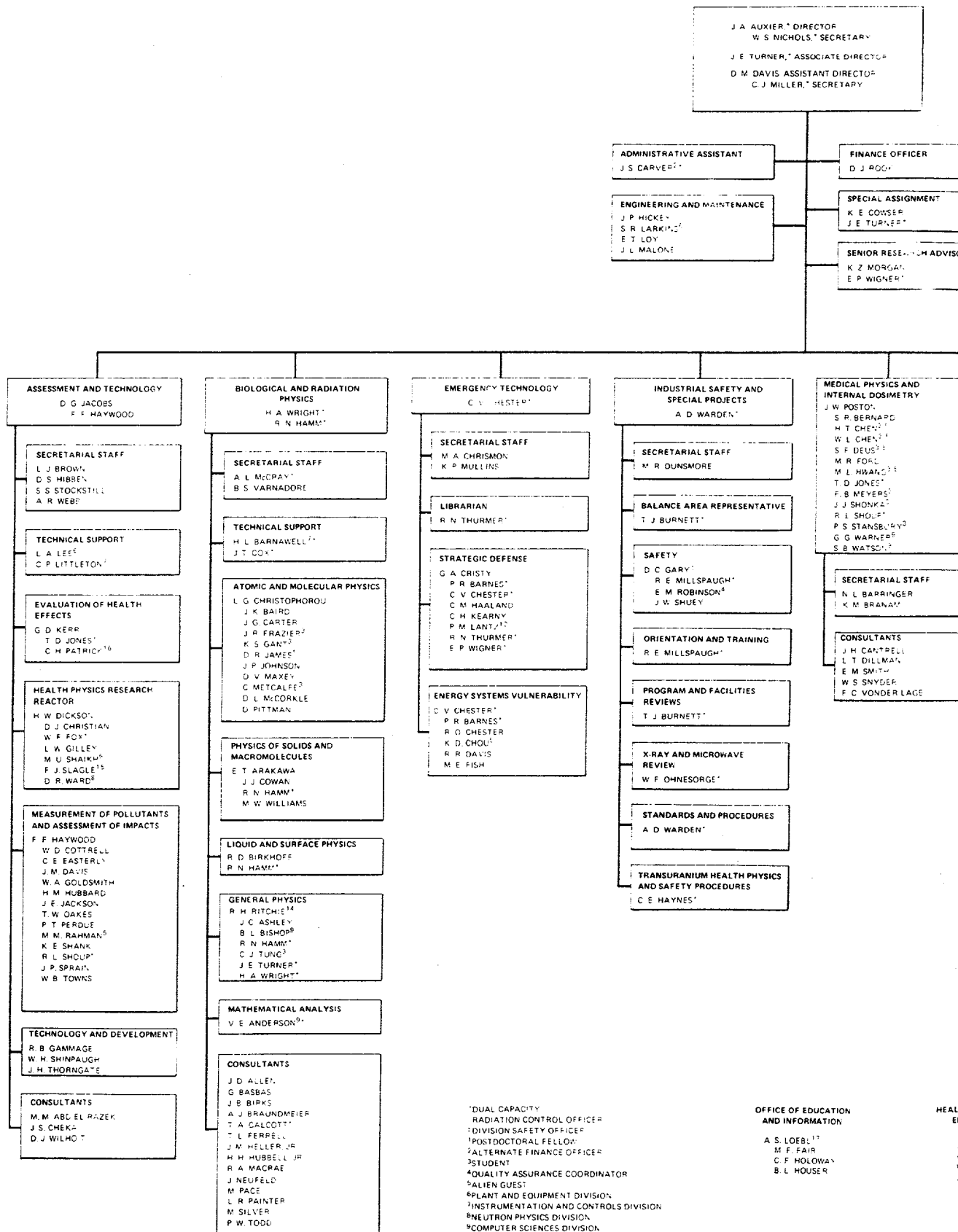
Radiation and Safety Surveys

R. L. Clark
H. M. Butler
R. E. Coleman
C. A. Golden
C. R. Guinn
L. C. Johnson
W. T. Martin
C. H. Miller
A. J. Smith
B. T. Walters
J. A. Worth

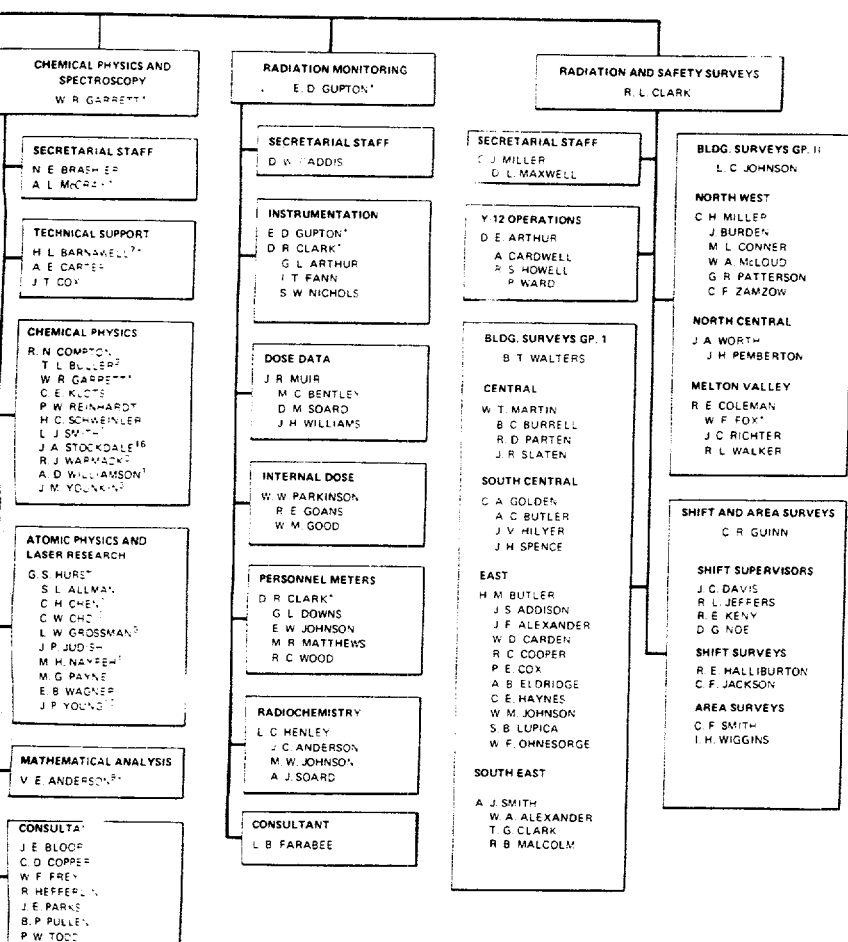
Industrial Safety and Special Projects

A. D. Warden
D. C. Gary

FEBRUARY 1 1973



1 DUAL CAPACITY
 2 RADIOATION CONTROL OFFICER
 3 DIVISION SAFETY OFFICER
 4 POSTDOCTORAL FELLOW
 5 ALTERNATE FINANCE OFFICER
 6 STUDENT
 7 QUALITY ASSURANCE COORDINATOR
 8 ALIEN GUEST
 9 PLANT AND EQUIPMENT DIVISION
 10 INSTRUMENTATION AND CONTROLS DIVISION
 11 NEUTRON PHYSICS DIVISION
 12 COMPUTER SCIENCES DIVISION
 13 ANALYTICAL CHEMISTRY DIVISION
 14 ORAU RESEARCH TRAINEE
 15 ON LOAN TO CHEMICAL TECHNOLOGY DIVISION
 16 ON LOAN TO FINANCE AND MATERIALS DIVISION
 17 SENIOR VISITING FELLOW, CAVENISH LABORATORY,
 UNIVERSITY OF CAMBRIDGE, ENGLAND
 18 UNDER R&D SUBCONTRACT
 19 LEAVE OF ABSENCE
 20 ENERGY DIVISION

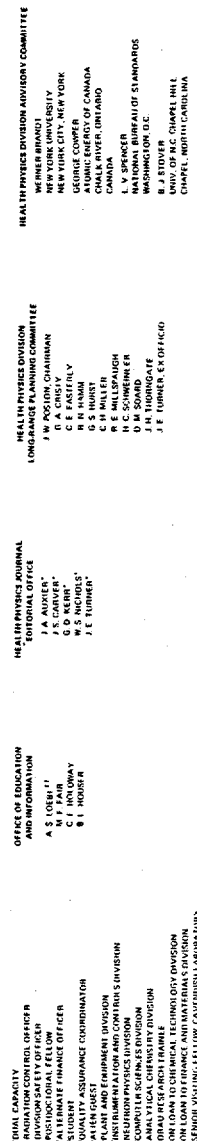


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HEALTH PHYSICS DIVISION
LONG-RANGE PLANNING COMMITTEE
J W POSTON CHAIRMAN
G A CRIST
C E EASTERLY
R N HAMM
G S HURST
C H MILLER
R E MILLSPAUGH
H C SCHWEINLER
D M SOARD
J H THORNGATE
J E TURNER EX OFFICIO

HEALTH PHYSICS DIVISION ADVISORY COMMITTEE
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WASHINGTON, D C
B J STOVER
UNIV. OF N.C. CHAPEL HILL
CHAPEL, NORTH CAROLINA

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2.0 SUMMARY

RADIATION MONITORING

Personnel Monitoring

There were no external or internal exposures to personnel which exceeded the standards for radiation protection as defined in ERDA Manual Chapter 0524. Only 71 employees received whole body radiation exposures greater than 1 rem. The highest whole body exposure dose equivalent to an employee was 2.7 rem. The highest internal exposure was less than one-half of the maximum permissible body burden.

Health Physics Instrumentation

During 1975, 36 portable instruments were added to the inventory and 28 retired. The total number in service on January 1, 1976, was 1,278. There were 16 facility radiation monitoring instruments installed and eight retired during 1975. The total number in service on January 1, 1976, was 971.

ENVIRONS MONITORING

Atmospheric Monitoring

There were no releases of gaseous waste from the Laboratory which were of a level that required an incident report to the ERDA. The average concentration of beta radioactivity in the atmosphere at the perimeter of the ERDA-controlled area was less than one percent of the value applicable to releases to uncontrolled areas.

Water Monitoring

There were no releases of liquid waste from the Laboratory which were of a level that required an incident report to the ERDA. The quantity of radionuclides of primary concern in the Clinch River averaged less than 0.5 percent of the MPC_w.

Radiation Background Measurements

The average background level at the PAM stations during 1975 was 8.7 $\mu\text{R/hr}$, or 76 mrem/yr.

Soil Samples

Nine soil samples were collected and analyzed for plutonium and uranium. Plutonium content ranged from $2.1 \times 10^{-8} \mu\text{Ci/g}$ to $5.4 \times 10^{-8} \mu\text{Ci/g}$, and the uranium content ranged from $26 \times 10^{-8} \mu\text{Ci/g}$ to $180 \times 10^{-8} \mu\text{Ci/g}$.

Calculation of Potential Radiation Dose to the Public

The maximum total body radiation dose to an individual residing continuously at the residence nearest ORNL was estimated to be 0.12 mrem/yr.

Quality Assurance Program

A Quality Assurance Program has been implemented in the Environs Monitoring Group.

RADIATION AND SAFETY SURVEYS

Laboratory Operations Monitoring

During 1975, the Radiation and Safety Surveys personnel continued to assist the operating groups in keeping the contamination, air concentration, and personnel exposure levels below the established maximum permissible levels. They assisted in reducing or eliminating a number of problems associated with radiation protection at the Laboratory.

Unusual Occurrences

Eleven unusual occurrences involving radioactive materials were recorded during 1975. The number reported for 1974 was 10, and the average number for the past five years (1971-1975) was 10.4.

Laundry Monitoring

Of the 537,000 articles of wearing apparel monitored during 1975, about 10 percent were found contaminated.

INDUSTRIAL SAFETY AND SPECIAL PROJECTS

Accident Analysis

There were two Disabling Injuries experienced at ORNL in 1975, a frequency rate of 0.27. The frequency rate for 1974 was 0.81. The Serious Injury frequency rate for 1975 was 2.25, as based on the new OSHA system for recording injuries and illness (RII).

Summary of Disabling Injuries

A total of 173 days were lost or charged for the two Disabling Injuries. One employee suffered a permanent-partial disability.

Safety Awards

The National Safety Council Award of Honor and the Union Carbide Corporation Award of Distinguished Safety Performance were earned by the Laboratory in 1975.

3.0 RADIATION MONITORING

3.1 Personnel Monitoring

All persons who enter Laboratory areas where there is a likelihood of exposure to radiation or radioactive materials are monitored for the kinds of exposure they are likely to sustain. External radiation dosimetry is accomplished mainly by means of badge-meters, pocket ion chambers, and hand exposure meters. Internal deposition is determined from bio-assays and in vivo counting.

3.1.1 Dose Analysis Summary, 1975

(a) External Exposures - No employee received a whole body radiation dose which exceeded the standards for radiation protection, ERDA Manual Chapter 0524. The maximum whole body dose sustained by an employee was about 2.7 rem or 23 percent of the maximum permissible annual dose. The range of doses to persons using ORNL badge-meters is shown in Table 3.1.1, page 9.

As of December 31, 1975, no employee had a cumulative whole body dose which was greater than the recommended maximum permissible value based on the age proration formula $5(N-18)$ (Table 3.1.2, page 9). No employee has an average annual dose that exceeded 5 rem per year of employment (Table 3.1.3, page 9). The greatest cumulative dose of whole body radiation received by an employee was approximately 97 rem. This dose was accrued over an employment period of about 29 years and represents an average dose of about 3.4 rem per year.

The greatest cumulative dose to the skin of the whole body received by an employee during 1975 was about 11.3 rem or 75 percent of the maximum permissible annual skin dose of 15 rem.

The maximum cumulative hand dose recorded during 1975 was about 22 rem or 29 percent of the recommended maximum permissible annual dose to the extremities.

The average of the 10 greatest whole body doses to ORNL employees for each of the years 1971 through 1975 is shown in Table 3.1.4, page 10. The maximum individual dose for each of those years is shown, also.

(b) Internal Exposures - One employee inhaled unknown compounds of ^{244}Cm resulting in an initial lung deposit of 25 to 50 percent of the maximum permissible lung burden. Pulmonary clearance was rapid and within 105 days the lung content had decreased to less than the minimum detectable activity (25 percent MPLB). Urinalysis results demonstrated that the systemic uptake was less than two percent of the maximum permissible body burden. Another employee sustained a wound to the left thumb, which (after excision) was found to contain less than one percent of the MPBB of ^{239}Pu . There was no case of internal exposure during the

year for which the radioactive material within the body averaged as much as one-half the maximum permissible organ burden for the year.

3.1.2 External Dose Techniques

(a) Badge Meters - Photobadge meters are issued to all employees and to non-employees who are authorized to have frequent access to ORNL facilities. Temporary meters are issued to casual visitors.

All badge-meters are equipped with nuclear accident metering devices and beta-gamma sensitive films. Various complements of TLD's, according to potential for radiation exposure, are included in photobadge meters. NTA films are included, also, in the badges of those who are likely to be exposed to fast neutrons.

Badge-meters of employees are routinely exchanged and processed each calendar quarter, or more frequently if required for exposure control. Meters issued to visitors are processed as may be required for monitoring purposes.

(b) Pocket Meters - Pocket meters (indirect reading, ionization chambers) are made available at all principal points of entry to ORNL premises. A pair of pocket meters is carried for the duration of a work shift by persons who work in an area where the potential for an exposure of 20 mR or more exists during the work shift. Pocket meter pairs are processed each day by Health Physics technicians, and readings of 20 mR or more are reported daily to supervision. Pocket meter readings are used for estimating integrated exposure and as a basis for badge-meter processing during a calendar quarter.

(c) Hand Exposure Meters - Hand exposure meters are TLD-loaded finger rings used to measure hand dose. Hand exposure meters are issued to persons for use during operations where it is likely that the hand dose may exceed 1 rem during the week. They are issued and collected by Radiation and Safety Surveys personnel who determine the need for this type of monitoring and arrange for a processing schedule.

(d) Metering Résumé - Shown in Table 3.1.5, page 11, are the quantities of personnel metering devices used and processed during 1975. The number of dosimeters processed is less than the number issued, because those which are issued for accident dosimetry only are not processed unless there is a likelihood of exposure.

3.1.3 Internal Dose Techniques

(a) Bio-Assay - Urine and fecal samples are analyzed for the purpose of making internal dose determinations. The frequency of sampling and the type of radiochemical analysis performed are based upon each specific radioisotope and the intake potential. Because of the small quantities of radioactive material in most samples, qualitative analyses are not feasible; and only quantitative analyses for predetermined isotopes are performed routinely.

In most cases, bio-assay data require interpretation to determine the dose to the person; computer programs are used for evaluation of extensive data on urinary excretion of ^{239}Pu . An estimate of dose is made for all cases in which it appears that one-fourth of a body burden, averaged over a calendar year, may be exceeded.

The analyses performed by the Applied Health Physics and Safety radiochemical lab during 1975 are summarized in Table 3.1.6, page 12.

(b) Whole Body Counter - The Whole Body Counter (an *in vivo* gamma spectrometer) may be used for estimating internally deposited quantities of most radionuclides which emit photons.

During calendar year 1975 there were 331 whole body, thorax or wound counts. In addition to the two cases described above [3.1.1(b)], there were four other cases of measurable activity detected. One had less than 15 percent of the maximum permissible lung burden (MPLB) of ^{233}U , another < 15 percent MPLB of ^{65}Zn and ^{57}Co , and another < 15 percent MPLB of ^{60}Co .

(c) Counting Facility - The Applied Health Physics and Safety counting facility determines radioactivity content of samples submitted by the applied sections. A summary of analyses is in Table 3.1.7, page 13.

3.1.4 Reports

Routine reports of personnel monitoring data are prepared and distributed to divisional supervision and to the Applied Health Physics and Safety staff.

(a) Pocket Meter Data - A report is prepared daily of the names, ORNL division, and readings for pocket meter readings which were 20 mR or greater during the previous 24 hours.

A computer-prepared report, which includes all pocket meter data for the previous week and summary data for the calendar quarter, is published and distributed weekly.

(b) External Dosimetry Data - A computer-prepared report, which includes data of recorded skin dose and whole body dose for the previous calendar quarter and totals for the current year, is published and distributed quarterly.

(c) Bio-Assay Data - A computer-prepared report, which includes data of sample status and results for the previous week, is published and distributed weekly. A quarterly and an annual report of results are prepared and distributed.

(d) Whole Body Counter Data - Preliminary results of analysis are reported on a card form soon after counting is done.

A computer-prepared report, which includes data collected during the previous calendar quarters of the calendar year, is published and distributed quarterly.

3.1.5 Records

Permanent records of personnel monitoring data are maintained for each person who is assigned an ORNL photobadge meter.

3.2 Health Physics Instrumentation

The Health Physics Division shares with the Instrumentation and Controls Division the responsibility for the selection of electronic radiation monitoring instruments used in the ORNL health physics program. Normally, the Health Physics Division is responsible for determining the need for new instrument types and modifications to existing types, for specifying the health physics requirements, and for approval of the design. The Health Physics Division is also responsible for calibrating all instruments used in the health physics program and is allocated the funds for maintenance of these instruments. Maintenance is performed or cross-ordered by the Instrumentation and Controls Division.

Non-electronic personnel monitoring devices are designed, tested, calibrated and maintained by Health Physics Division personnel.

3.2.1 Instrument Inventory

The electronic instruments used in the health physics program are divided, for convenience in servicing and calibrating, into two classes: the first class includes battery-powered portable instruments; the second class includes the stationary instruments that are AC powered. Portable instruments are assigned and issued to the Radiation and Safety Surveys Complexes. Stationary instruments are the property of the ORNL division which has the monitoring responsibility in the area in which the instrument is located. Table 3.2.1, page 14, lists portable instruments assigned at the end of 1975; Table 3.2.2, page 14, lists stationary instruments at the X-10 site in use at the end of 1975.

Inventory and Service Summaries for health physics instruments are prepared on an IBM 360. These computer-programmed reports enable the Instruments Group to maintain a current inventory on most health physics instrument requirements.

The allocation of stationary health physics monitoring instruments at the X-10 site by division is shown in Table 3.2.3, page 15.

3.2.2 Calibration Facility

The Health Physics Division maintains a calibration facility for the calibration and maintenance of portable radiation instruments and personnel

metering devices. The facility is equipped with calibration sources, remote control devices, and shop space for the use of Instrumentation and Controls Division maintenance personnel. Health Physics personnel assign, arrange for maintenance of, calibrate, provide delivery services for, and maintain inventory and servicing data of all portable health physics instruments.

Portable instruments should be serviced (1) whenever repairs are needed, (2) at least once each two months for those which have replacement-type batteries, and (3) at least once each three months for those instruments which have "permanent" (rechargeable) batteries. The number of calibrations of portable instruments for 1975 is shown in Table 3.2.4, page 16.

3.3 Developments

3.3.1 Bio-Assay

The urinalysis data for the two employees who inhaled ^{244}Cm in 1974 were found to follow closely the time dependence of ^{241}Am retention observed by Cohen and Wrenn.¹ Accordingly, the excretion data for the employees was fitted with a series of three exponentials having the half-times reported by Cohen and Wrenn. Conformance to the fitted equations was good and estimates of initial intakes appeared plausible. The two employees were calculated to have initial uptakes of 7 and 16 nCi of ^{244}Cm , respectively.² These systemic burdens are about half the estimated uptakes made shortly after exposure.

3.3.2 Whole Body Counter

Since a number of workers at this Laboratory have occasion to handle insoluble oxides of the transuranic elements, it has been necessary to develop improved instrumental and analytical methods to evaluate lung burdens following accidental inhalation of ^{239}Pu , ^{241}Am , ^{244}Cm and other isotopes. As part of a continuing effort to improve our instrumental techniques, we have obtained correlated time and energy spectra for the ORNL phoswich system in order to establish and control the most critical instrumental parameters. From the results obtained, procedures have been derived for establishing and maintaining the phoswich's time gate. In addition, we have shown that the effectiveness of the phoswich's background cancellation diminishes with pulse height and may also, like resolution, deteriorate with time. From these experiments, it became obvious that our crystal assembly had suffered deterioration and a new phoswich unit has, therefore been ordered so that adequate personnel monitoring can be maintained.

¹N. Cohen and M. E. Wrenn, Radiation Res. 55, 129 (1973).

²W. W. Parkinson, L. C. Henley, R. E. Goans and W. M. Good, "Evaluation of Two Cases of ^{244}Cm Inhalation," Proceedings of the Ninth Midyear Topical Symposium of the Health Physics Society, p. 582 (1976), Denver, Colorado.

Recent low-level counting experience at the Whole Body Counter has also demonstrated that the superior energy resolution of semiconductor radiation detectors gives these devices a significant advantage over state-of-the-art phoswich assemblies for in vivo monitoring of actinides. In line with our efforts to improve personnel monitoring techniques, a joint effort was recently initiated between the Chemistry and Health Physics Divisions at ORNL to investigate use of large-area, high-resolution planar germanium detectors for analysis of respired actinide nuclides. Using the largest intrinsic Ge detector available (10 cm² x 1.2 cm thick) we were able to measure L series X-rays in humans and in phantoms with ~ 400 eV resolution and with essentially geometrical efficiency as detectors of this thickness are "black" to photons up to ~ 130 keV. Results from these joint experiments indicate that significantly improved sensitivities for detection of Pu will be realized through multiplexing of Ge arrays.

In view of these developments, a multi-divisional effort involving the transuranium chemistry, internal dosimetry and radiation monitoring groups is now well established to bring advanced technology to bear on the problem of determining lung burdens resulting from occupational exposure. Extensive Monte Carlo calculations are also currently being pursued to determine the spatial distribution of X-ray photons under different lung distributions of ²³⁹Pu. From our current study, we expect to acquire sufficient information from the modeling of X-ray transport in the human body and from measurements on humans to determine design parameters for an intrinsic germanium array. We feel that a substantial increase in sensitivity can be achieved through multiplexing of several semiconductor detectors, but the problem of in vivo actinide detection is of sufficient complexity to warrant a careful parametric study of design possibilities before large sums of money are spent by various laboratories for systems that may do little or nothing to attain lower detectable levels.

In conclusion, from these studies we have been able to obtain much improved calibration factors for various actinides as well as understand certain features of the natural room background which were unresolvable with NaI detectors. Computer codes have also been developed to analyze these data as well as personnel data taken from our phoswich operation. With improved calibration factors, with improved and more sophisticated electronics and with elegant spectral stripping and correlation algorithms, we are confident that a reduction in the detection limit for ²³⁹Pu can be affected.

Table 3.1.1 Dose Data Summary for Laboratory Population
Involving Exposure to Whole Body Radiation—1975

Group	Number of Rem Doses in Each Range							Total
	0-0.1	0.1-1	1-2	2-3	3-4	4-5	5 up	
ORNL Employees	4906	465	58	13	0	0	0	5442
ORNL-Monitored Non-Employees	91	17	0	0	0	0	0	108
TOTAL	4997	482	58	13	0	0	0	5550

Table 3.1.2 Average Rem Per Year Since Age 18—1975

Group	Number of Doses in Each Range				Total
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	5438	4	0	0	5442

Table 3.1.3 Average Rem Per Year of Employment at ORNL—1975

Group	Number of Doses in Each Range				Total
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	5432	10	0	0	5442

Table 3.1.4 Average of the Ten Highest Whole Body
Doses and the Highest Individual Dose by Year

Year	Average of the Ten Highest Doses (Rem)	The Highest Dose (Rem)
1971	3.41	4.95
1972	4.18	4.88
1973	3.12	4.63
1974	2.34	3.58
1975	2.41	2.71

Table 3.1.5 Personnel Meters Services

	<u>1973</u>	<u>1974</u>	<u>1975</u>
A. Pocket Meter Usage			
1. Number of Pairs Used			
ORNL	75,888	84,864	80,860
CPFF	<u>6,032</u>	<u>10,452</u>	<u>9,984</u>
Total	81,920	95,316	90,844
2. Average Number of Users per Quarter			
ORNL	753	804	806
CPFF	<u>96</u>	<u>160</u>	<u>146</u>
Total	849	964	952
B. Meters Processed for Monitoring Data			
1. Beta-Gamma Badge-Meter	19,430	18,490	23,600
2. Neutron Badge-Meter	1,390	550	680
3. Hand Meter	1,400	810	670

Table 3.1.6 Radiochemical Lab Analyses—1975

Radionuclide	Urine	Feces	Milk	Sediment and Soil	Water	Controls
Plutonium, Alpha	502	6		38	128	95
Transplutonium, Alpha	397	5		15	128	70
Uranium, Alpha	236	5		13		10
Strontium, Beta	315		468		12	30
Cesium-137	16					10
Tritium	93				140	35
Iodine-131			468			
Other	113					5
TOTALS	1672	16	936	58	280	255

Table 3.1.7 Counting Facility Analyses—1975

Types of Samples	Number of Samples			Unit Total
	Alpha	Beta	Gamma	
Facility Monitoring				
Smears	46,603	48,608		95,211
Air Filters	13,998	11,980		25,978
Environs Monitoring				
Air Filters	3,142	3,142	250	6,534
Fallout		2,986	100	3,086
Rainwater		761		761
Surface Water		156		156
Milk		468		468

Table 3.2.1 Portable Instrument Inventory—1975

Instrument Type	Instruments Added 1975	Instruments Retired 1975	In Service Jan. 1, 1976
G-M Survey Meter	17	7	448
Cutie Pie	13	6	421
Alpha Survey Meter	2	7	279
Neutron Survey Meter	3	0	107
Miscellaneous	1	8	23
TOTAL	36	28	1,278

Table 3.2.2 Inventory of Facility Radiation Monitoring
Instruments for the Year—1975

Instrument Type	Installed During 1975	Retired During 1975	Total Jan. 1, 1976
Air Monitor, Alpha	2	0	104
Air Monitor, Beta	0	3	168
Lab Monitor, Alpha	10	0	177
Lab Monitor, Beta	2	0	206
Monitron	0	1	206
Other	2	4	110
TOTAL	16	8	971

Table 3.2.3 Health Physics Facility Monitoring Instruments
Divisional Allocation at X-10 Site—1975

ORNL Division	α Air Monitor	β Air Monitor	α Lab Monitor	β Lab Monitor	Monitron	Other	Total
Analytical Chemistry	7	13	16	19	15	5	75
Chemical Technology	48	52	63	33	40	32	268
Chemistry	8	9	19	24	19	7	86
Metals and Ceramics	11	6	14	4	5	10	50
Isotopes*	18	28	33	45	51	19	194
Operations	2	49	9	31	57	15	163
All Others	10	11	23	50	19	22	135
TOTAL	104	168	177	206	206	110	971

*Divisional reallocation under way.

Table 3.2.4 Calibrations Facility Résumé—1975

	1975
Beta-Gamma	2,712
Neutron	318
Alpha	971
Personal Dosimeters	1,810
Badge Dosimetry Components	15,400

4.0 ENVIRONMENTAL MONITORING

The Health Physics Division monitors for airborne radioactivity in the East Tennessee area by the use of three separate monitoring networks. The local air monitoring (LAM) network (Figs. 4.0.1, page 25, and 4.0.2, page 26) consists of 22 stations that are positioned relatively close to ORNL operational activities; the perimeter air monitoring (PAM) network (Fig. 4.0.3, page 27) consists of nine stations located on the perimeter of the ERDA-controlled area and provides data for evaluating the impact of all Oak Ridge operations on the immediate environment; and the remote air monitoring (RAM) network (Fig. 4.0.4, page 28) consists of eight stations located outside the ERDA-controlled area at distances of from 12 to 75 miles from ORNL. The monitoring networks provide for the collection of (1) airborne radioactivity by air filtration techniques, (2) radioparticulate fallout material by impingement on gummed paper trays, (3) rainwater for measurement of fallout occurring as rainout, and (4) radioiodine using charcoal cartridges.

Low-level radioactive liquid wastes originating from ORNL operations are discharged, after treatment, to White Oak Creek, which is a small tributary of the Clinch River. The radioactive content of White Oak Creek discharge is determined at White Oak Dam, which is the last control point along the stream prior to the entry of White Oak Creek into the Clinch River. Water samples are collected at several locations in the Clinch River, beginning at a point above the entry of the wastes into the river and ending at Center's Ferry near Kingston, Tennessee, the nearest population center downstream (Fig. 4.0.5, page 29).

Samples of White Oak Creek effluent are collected at White Oak Dam by a continuous proportional sampler and analyzed weekly for gross beta activity as a control measure and as a means of evaluating the gross concentration of radioactivity entering the Clinch River. Portions of the weekly samples are composited into monthly samples for detailed analyses by gamma spectrometric and wet-chemical techniques. The weekly samples are analyzed for transuranic alpha emitters, total strontium, tritium, and iodine-131. The monthly composites are concentrated and analyzed by radiochemical and gamma spectrometric techniques, normally for the following: strontium-90, cesium-137, barium-140, cerium-144, ruthenium-106, zirconium-95, niobium-95, cobalt-60, tritium, plutonium, transplutonium, and gross beta. Calculations are made of the concentrations of radioactivity in the Clinch River at the point of entry of White Oak Creek, using the concentrations measured at White Oak Dam and the dilution provided by the river. To verify the calculated concentrations, two sampling stations are maintained in the Clinch River below the point of entry of the wastes; one at the Oak Ridge Gaseous Diffusion Plant (ORGDP) water intake (Clinch River Mile [CRM] 14.5) and the other at Center's Ferry near Kingston, Tennessee (CRM 4.5). In addition, a sampling station is maintained in the Clinch River above the point of entry of the waste at Melton Hill Dam (CRM 23.1) to provide background data.

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The ORGDP water sampling station collects a sample from the Clinch River proportional to the flow in the river near the water intake of the ORGDP water system. The samples are brought into the Laboratory at weekly intervals, and an aliquot is composited for quarterly analysis of tritium. The remaining portion of the sample is passed over anion and cation resins to remove nuclides. At quarterly intervals, the resin columns are eluted, and the eluent is analyzed for gross activity and for individual radio-nuclides that may be present in significant amounts.

A "grab" sample is collected daily at the Center's Ferry sampling station which is located on the Clinch River at CRM 4.5. The daily grab samples are composited and analyzed on a quarterly basis. The preparation of these samples and the analyses performed are the same as those for the ORGDP water sampling station.

The Melton Hill Dam sampling station collects a sample proportional to the flow of water through the power-generating turbines, which represents all of the discharge from the Dam other than a minor amount discharged in the operation of the locks. Samples are collected from the station at weekly intervals, processed, and analyzed in the same manner as for the ORGDP water sampling station.

Samples of ORNL potable water are collected daily, composited, and stored. At the end of each quarter, these composites are analyzed radio-chemically for ^{90}Sr content and are assayed for long-lived gamma-emitting radionuclides by gamma spectrometry.

Raw milk is collected at 11 sampling stations located within a radius of 50 miles from ORNL. Samples are taken on a weekly basis from seven stations (Fig. 4.0.6, page 30) located outside the ERDA-controlled area within a 20-mile radius of ORNL. Samples are collected every five weeks from the four remaining stations (Fig. 4.0.7, page 31) located more remotely with respect to Oak Ridge operations out to distances of about 50 miles. The purpose of the milk sampling program is twofold: first, samples collected in the immediate vicinity of ORNL provide data by which one may evaluate the possible effect of effluents from ORNL operations; second, samples collected remote to the immediate vicinity of ORNL provide background data which are essential in establishing a proper index from which releases of radioactive materials originating from Oak Ridge operations may be evaluated. The milk samples are analyzed by radiochemical techniques for strontium-90 and iodine-131. The minimum detectable concentrations of strontium-90 and iodine-131 in milk are 0.5 pCi/liter and 0.45 pCi/liter, respectively.

External gamma radiation background measurements are made routinely at the nine perimeter air monitoring stations, at one station located near Melton Hill Dam and at the remote monitoring stations; measurements are made using calcium fluoride thermoluminescent dosimeters suspended one meter above the ground. Dosimeters at the perimeter stations and Melton Hill Dam are collected and analyzed monthly. Those at remote stations are collected and analyzed semiannually.

External gamma radiation measurements are also made routinely along the bank of the Clinch River from the mouth of White Oak Creek several hundred yards downstream (Fig. 4.0.8, page 32). These measurements were used to evaluate gamma radiation levels resulting from ORNL liquid effluent releases and "sky shine" from an experimental ^{137}Cs plot located near the river bank. Radiation measurements were made using lithium fluoride thermoluminescent dosimeters suspended one meter above the ground surface.

Two species of fish which are commonly caught and eaten—white crappie and carp—are taken from the Clinch River during the spring and summer of each year. The fish are prepared for radiochemical analysis in a manner analogous to that for human consumption. Ten fish of each species are composited for each sample, and the samples are analyzed by gamma spectrometric and radiochemical techniques for the critical radionuclides which may contribute significantly to the potential radiation dose to man.

Soil samples are collected annually from locations near the PAM stations. Nine samples, approximately three inches in diameter and one centimeter thick, are collected in a one-square-meter area at each location, composited, and analyzed radiochemically for uranium and plutonium to determine background information for future comparison in the event of an accidental release.

4.1 Atmospheric Monitoring

4.1.1 Air Concentrations

The average concentrations of beta radioactivity in the atmosphere, as measured with filters from the LAM, PAM, and RAM networks during 1975, were as follows:

<u>Network</u>	<u>Concentration ($\mu\text{Ci/cc}$)</u>
LAM	5.8×10^{-14}
PAM	4.0×10^{-14}
RAM	4.2×10^{-14}

The LAM network value of $5.8 \times 10^{-14} \mu\text{Ci/cc}$ is less than 0.002 percent of the MPCU_a ¹ based on occupational exposure of $3 \times 10^{-9} \mu\text{Ci/cc}$. Both the PAM and RAM network values represent < 0.05 percent of the MPCU_a of $1 \times 10^{-10} \mu\text{Ci/cc}$ applicable to releases to uncontrolled areas. A tabulation of data for each station in each network is given in Table 4.1.1, page 33. The weekly values for each network are illustrated in Table 4.1.2, page 34.

¹The MPCU_a is defined as the maximum permissible concentration for an unknown mixture of radioisotopes in air. ERDA Manual Chapter 0524, Appendix, Annex 1, gives exposure values applicable to various mixtures of radionuclides and establishes guidelines for deriving the MPCU_a .

The values measured for 1975 are lower than those for 1974 by a factor of ~ 2 for the LAM, PAM, and RAM networks.

4.1.2 Fallout (Gummed Paper Technique)

The average activity from radioparticulate fallout measured for 1975 was lower than those for 1974 by a factor of ~ 3 for the LAM, PAM, and RAM networks. The average activity and number of particles per square foot are shown in Table 4.1.3, page 35.

4.1.3 Rainout (Gross Analysis of Rainwater)

The average concentration of beta radioactivity in rainwater collected from the three networks during 1975 was as follows:

<u>Network</u>	<u>Concentration ($\mu\text{Ci/cc}$)</u>
LAM	1.6×10^{-8}
PAM	1.5×10^{-8}
RAM	2.4×10^{-8}

The average concentration of beta radioactivity measured for 1975 was lower than those for 1974 by a factor of ~ 2 for the three networks. The average concentration measured at each station within each network is presented in Table 4.1.4, page 36. The average concentration for each network for each week is given in Table 4.1.5, page 37.

4.1.4 Atmospheric Radioiodine (Charcoal Cartridge Technique)

Atmospheric iodine sampled at the perimeter stations averaged $0.7 \times 10^{-14} \mu\text{Ci/cc}$ during 1975. This average represents < 0.01 percent of the maximum permissible concentration of $1 \times 10^{-10} \mu\text{Ci/cc}$ applicable to inhalation of ^{131}I released to uncontrolled areas. The maximum concentration observed at any one station for one week was $2.7 \times 10^{-14} \mu\text{Ci/cc}$ at PAM 31, the perimeter station located at the Kerr Hollow Gate.

The average radioiodine concentration at the local stations was $2.5 \times 10^{-14} \mu\text{Ci/cc}$. This concentration is less than 0.01 percent of the maximum permissible concentration for inhalation by occupational personnel. The maximum concentration at any one station for one week was $17 \times 10^{-14} \mu\text{Ci/cc}$ at LAM 6 located southwest of Building 3027.

Table 4.1.6, page 38, presents the ^{131}I weekly average concentration data for both the local area (LAM) and the perimeter area (PAM) air monitoring networks. The weekly average ^{131}I concentration in air measured by stations in the LAM and PAM networks is given in Table 4.1.7, page 39.

The results of the specific radionuclide analyses of the filters from the three networks are given in Table 4.1.8, page 40.

4.1.5 Milk Analysis

The quarterly average and maximum concentrations of ^{131}I and ^{90}Sr in raw milk are given in Tables 4.1.9, page 41, and 4.1.10, page 42. If one assumes the average intake of milk per individual to be one liter per day, the concentrations of ^{131}I in milk collected near ORNL and in milk collected more remotely from ORNL are within FRC Range I. The concentrations of ^{90}Sr in milk from both the immediate and remote environs of ORNL are also within FRC Range I.

The concentration of ^{90}Sr in milk varies with locations; part of the variation has been found to result from differences in farming methods. Pastureland that is not fertilized and is overgrazed (a not too uncommon practice in this area) apparently results in a higher than normal concentration of ^{90}Sr in milk from cows pastured on this land.

4.1.6 ORNL Stack Releases

The ^{131}I releases from ORNL stacks are summarized in Table 4.1.11, page 43.

4.2 Water Monitoring

4.2.1 White Oak Lake Waters

Yearly discharges of specific radionuclides to the Clinch River, 1968 through 1975, are shown in Table 4.2.1, page 44.

The calculated average concentrations of the significant radionuclides in the Clinch River at Clinch River Mile (CRM) 20.8 (the point of entry of White Oak Creek into the river) are presented in Table 4.2.2, page 45. The concentration did not exceed one percent of MPC_w for any month during 1975 (Table 4.2.3, page 46).

The annual average percent MPC_w of beta emitters, other than tritium in the Clinch River, 1968 through 1975, is given in Table 4.2.4, page 47. Table 4.2.5, page 48, lists the annual average percent MPC_w of tritium in the Clinch River, 1968 through 1975.

4.2.2 Clinch River Water

The measured average concentrations and the percent of MPC_w of radionuclides in the Clinch River at Melton Hill Dam (CRM 23.1), about three miles upstream, at Gallaher (CRM 14.5), about six miles downstream, and at Center's Ferry (CRM 4.5, about 16 miles downstream from the entry of White Oak Creek, are given in Table 4.2.2, page 45.

4.2.3 Potable Water

The average concentrations of ^{90}Sr in potable water at ORNL during 1975 were as follows:

<u>Quarter Number</u>	<u>Concentration of ^{90}Sr ($\mu\text{Ci/ml}$)</u>
1	5.0×10^{-11}
2	5.0×10^{-11}
3	5.0×10^{-11}
4	9.0×10^{-11}
Average for Year	6.0×10^{-11}

The average value of 6.0×10^{-11} represents < 0.1 percent of the MPC_w for drinking water applicable to individuals in the general population.

4.2.4 Radionuclides in Clinch River Fish

The results of the analysis of fish samples are tabulated in pCi/kg of wet weight (Table 4.2.6, page 49) for each radionuclide of significance. An estimate of man's intake of radionuclides from eating the fish is made by assuming an annual rate of fish consumption of 37 pounds. An estimated percentage of maximum permissible intake is calculated by assuming a maximum permissible intake of fish to be comparable to a daily intake of 2.2 liters of water containing the MPC_w of these radionuclides for a period of one year.

4.3 Radiation Background Measurements

Data on the average external gamma radiation background rates are given in Table 4.3.1, page 50. The slight difference between the average levels in the perimeter and remote environs is considered to be within the variation in background levels normally experienced in East Tennessee which is dependent upon elevation, topography, and geological character of the surrounding soil.²

The average external gamma radiation levels along the bank of the Clinch River adjacent to an experimental cesium field are given in Table 4.3.2, page 51.

4.4 Soil Samples

Data on uranium and plutonium concentrations in soil samples are given in Table 4.4.1, page 52.

4.5 Environmental Monitoring Samples

A listing of environmental monitoring samples processed by type sample, type of analyses, and number of samples is given in Table 4.5.1, page 53.

²W. M. Lowder et al., Indoor Radon Daughter and Radiation Measurements in East Tennessee and Central Florida, HASL TM 71-8, March, 1971.

4.6 Calculation of Potential Radiation Dose to the Public

To determine the radiation doses resulting from the gaseous discharges from ORNL, the Gaussian plume model developed by Pasquill³ and Gifford⁴ was incorporated into a computer program. The effluents were assumed to occur from a single 300-foot stack, and meteorological data collected at the ORNL site were used.

The maximum total body radiation dose to an individual residing continuously at the residence nearest ORNL (which is ~ 4300 meters in the southwest direction) was estimated to be 0.12 mrem/yr, or .02 percent of the allowable standard. The principal contributing radionuclide to this dose is ¹³³Xe.

The average total body dose to an Oak Ridge resident was calculated to be .05 mrem/hr, which is .01 percent of the allowable standard. The cumulative whole body dose to the population within a 50-mile radius of ORNL, resulting from 1975 atmospheric effluents, was calculated to be 6.6 man-rem. This dose may be compared to an estimated 74,000 man-rem to the same population from natural background radiation.

The point of maximum potential exposure to an individual on the site boundary is located along the bank of the Clinch River adjacent to an experimental cesium field and is due primarily to "sky shine" from the field. A maximum potential whole body exposure of 310 mrem/yr was calculated for this location, assuming that an individual remained at this point for 24 hours/day for the entire year. The calculated maximum potential exposure is 62 percent of the allowable standard.⁵ This is an atypical exposure location, and the probability of an exposure of the magnitude calculated is considered remote since access is only by boat.

The total body dose to a "hypothetical maximum exposed individual" at the same location was calculated, using a more realistic residence time of 240 hours/yr. The calculated dose under these conditions was 8.4 mrem/yr, which is 1.7 percent of the allowable standard and represents what is considered a probable upper limit of exposure.

4.7 Quality Assurance Program

During the fall of 1975, a Quality Assurance Program was implemented in the Environmental Monitoring Group. In this program, a document is being prepared containing detailed descriptions of the type of measurements, sampling techniques, equipment and instrumentation, and analytical techniques used in the Environmental Monitoring Program.

³F. Pasquill, Atmospheric Diffusion, D. Van Nostrand Co., Ltd., London, 1962.

⁴F. A. Gifford, Jr., The Problem of Forecasting Dispersion in the Lower Atmosphere, U.S.A.E.C., D.T.I., 1962.

⁵ERDA Manual Chapter 0524.

A plan and procedure for auditing the application of all quality assurance plans will be developed. A system will also be established that will ensure that conditions adverse to these plans, such as defective equipment, techniques, etc., can be promptly identified and corrected.

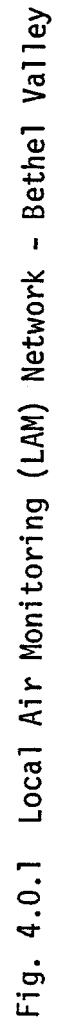


Fig. 4.0.1 Local Air Monitoring (LAM) Network - Bethel Valley

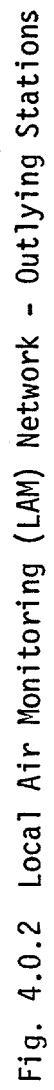


Fig. 4.0.2 Local Air Monitoring (LAM) Network - Outlying Stations

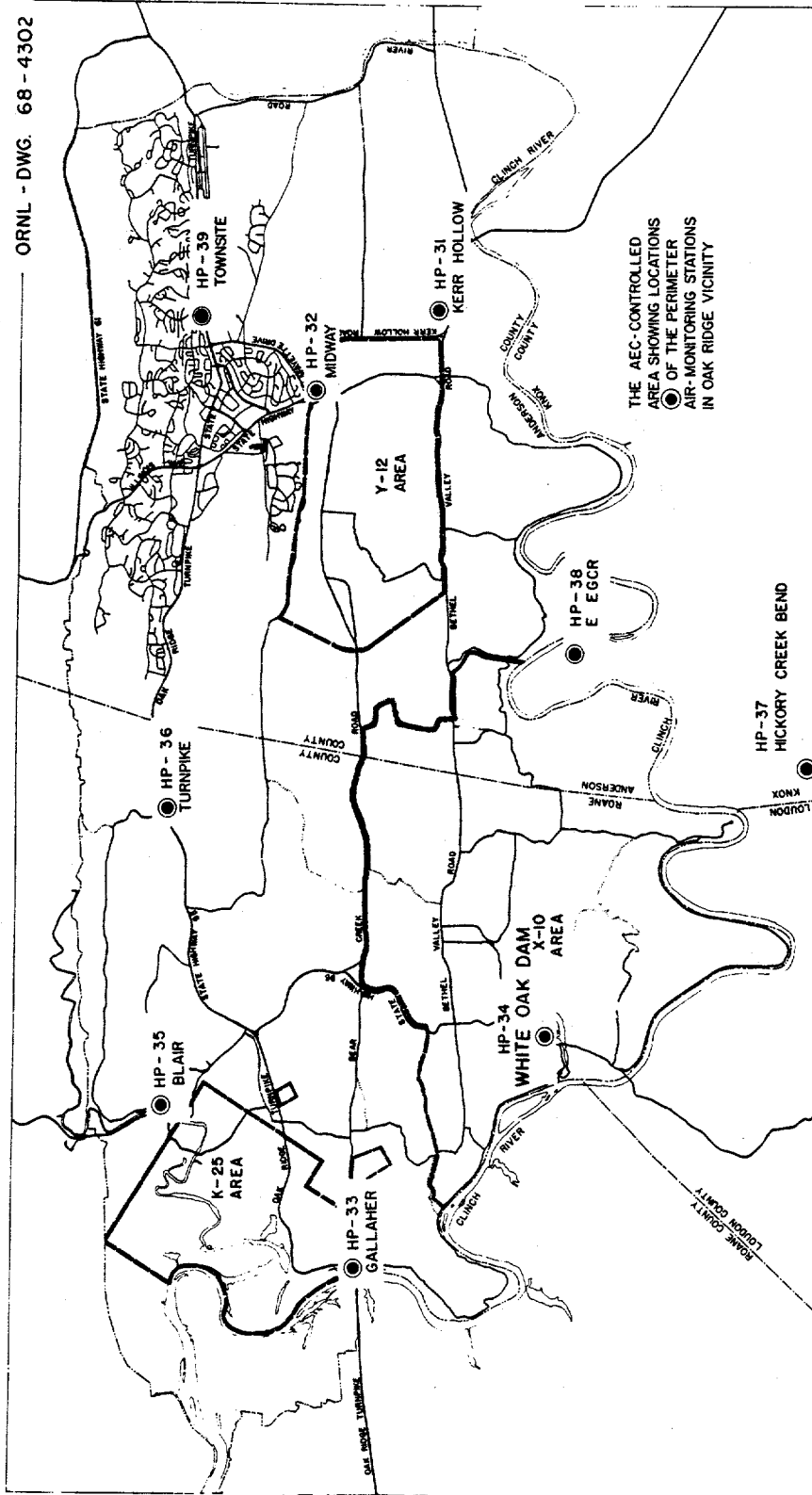


Fig. 4.0.3 Perimeter Air Monitoring (PAM) Network

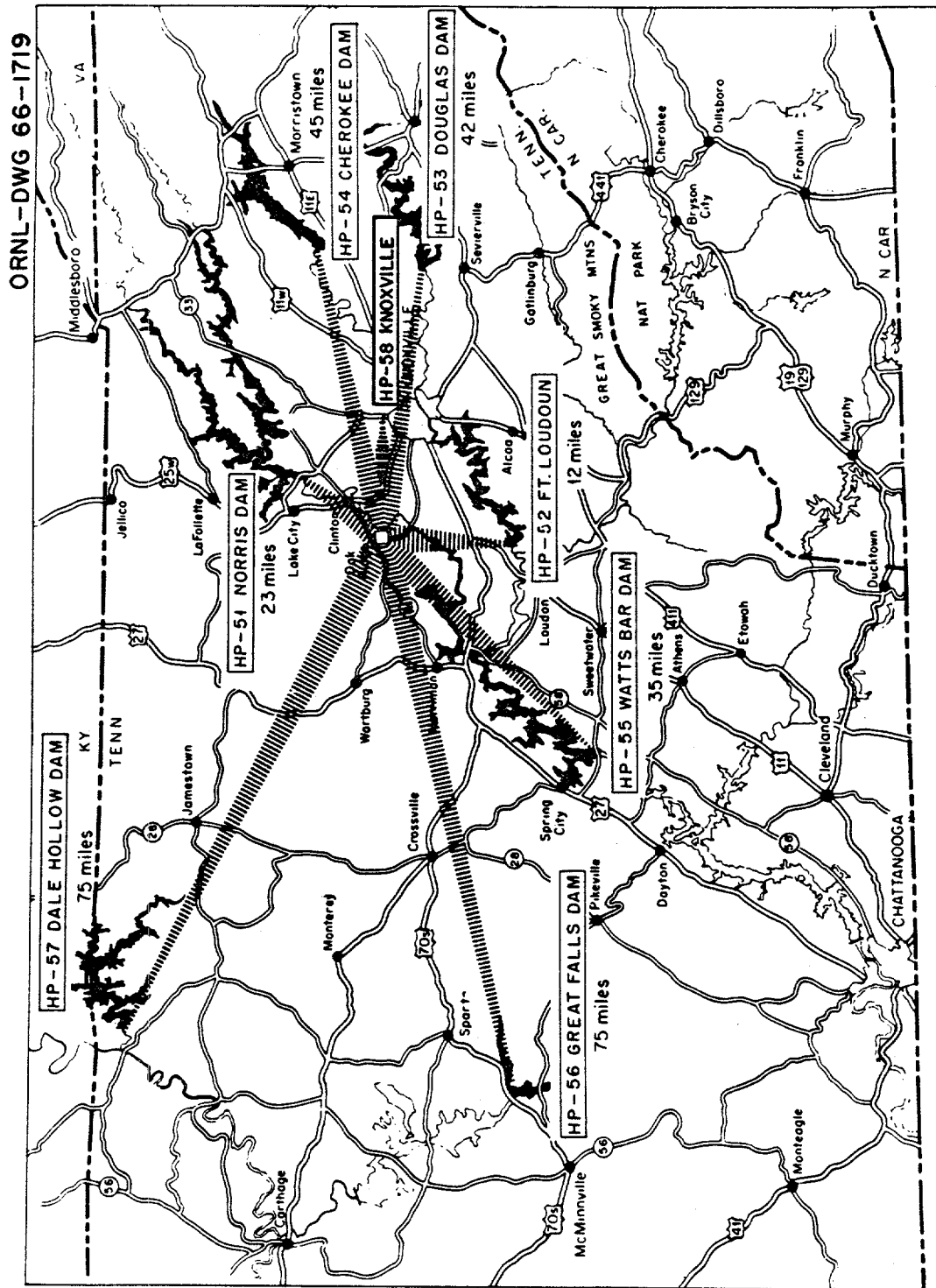


Fig. 4.0.4 Remote Air Monitoring (RAM) Network

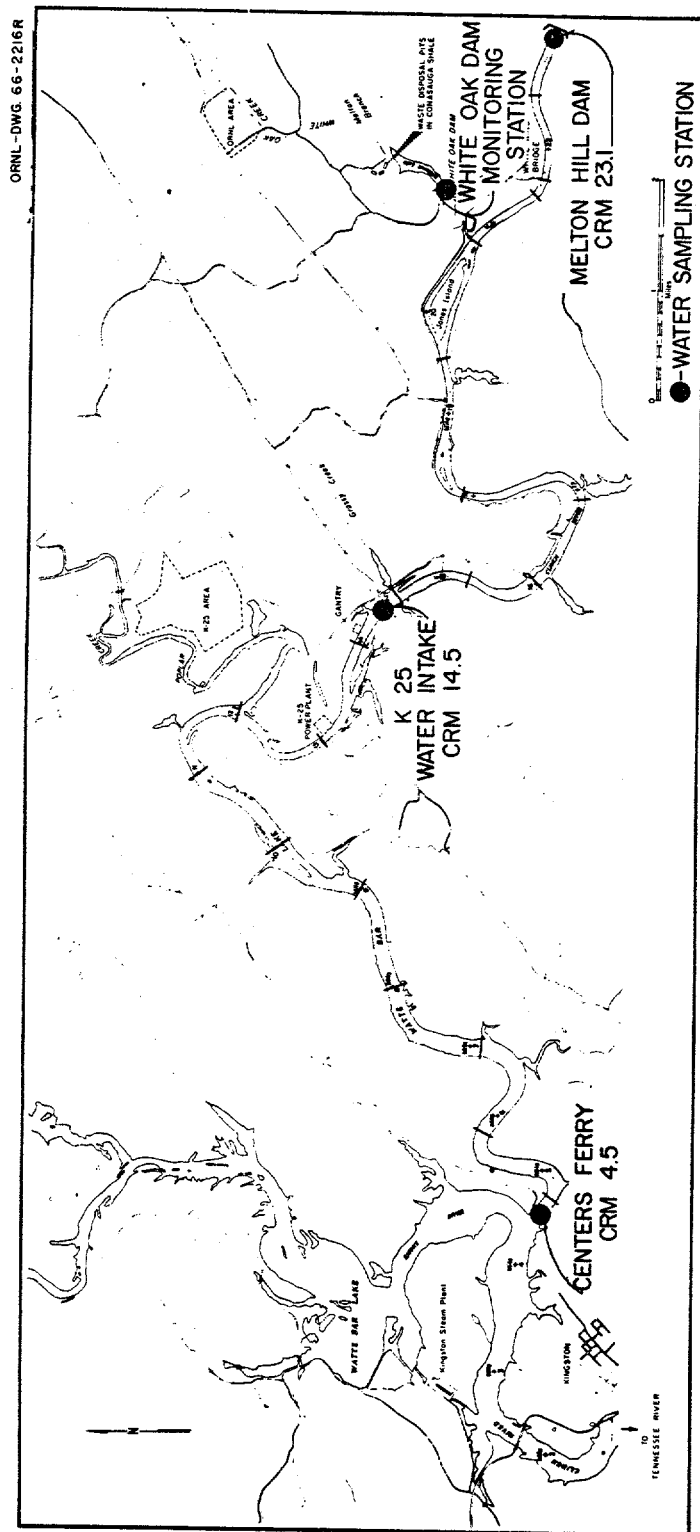


Fig. 4.0.5 Map Showing Water Sampling Locations in the East Tennessee Area



Fig. 4.0.6 Location of Milk Sampling Stations
(Within 20-Mile Radius of ORNL)

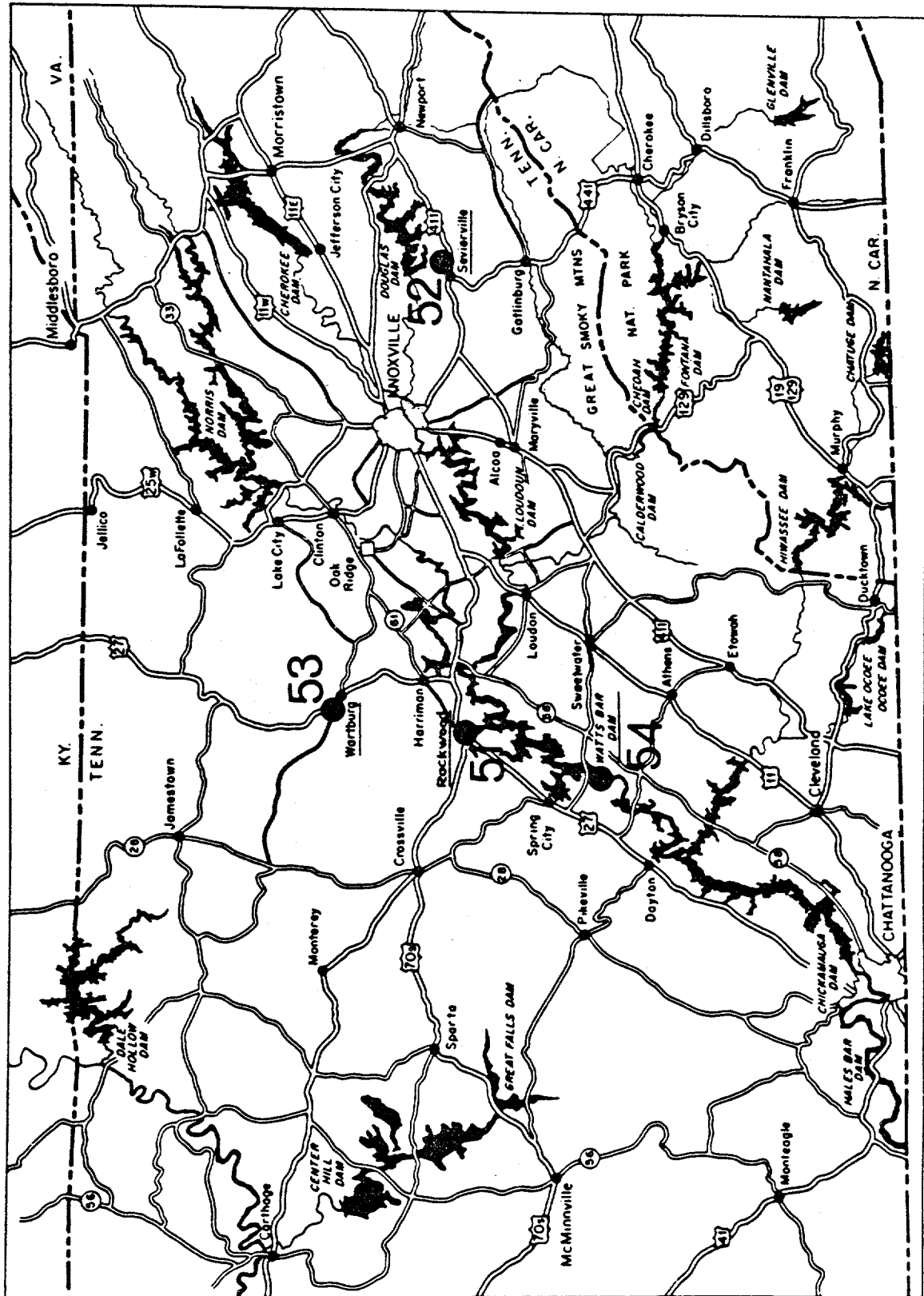


Fig. 4.0.7 Remote Environs Milk Sampling Locations

Table 4.1.2 Concentration of Beta Radioactivity in Air
as Determined from Filter Paper Data—1975
(System Average—by Weeks)

Week Number	Units of 10^{-14} $\mu\text{Ci/cc}$			Week Number	Units of 10^{-14} $\mu\text{Ci/cc}$		
	LAM's	PAM's	RAM's		LAM's	PAM's	RAM's
1	6.7	5.4	4.8	29	3.0	1.7	2.2
2	7.9	5.2	5.9	30	3.4	3.0	2.6
3	8.2	5.5	5.9	31	2.5	1.1	2.0
4	7.9	5.7	5.5	32	1.9	1.2	1.0
5	8.9	7.4	7.4	33	2.3	2.3	2.0
6	7.8	5.0	8.4	34	4.2	3.8	2.7
7	11	8.0	7.6	35	2.2	2.0	1.4
8	10	8.5	7.5	36	2.8	2.5	2.5
9	11	7.6	7.3	37	1.8	1.5	1.6
10	9.4	6.6	8.7	38	1.7	1.2	1.4
11	6.2	3.9	4.8	39	1.1	1.0	1.2
12	10	6.7	6.8	40	2.4	2.1	1.7
13	8.7	6.4	6.1	41	3.6	2.5	2.3
14	11	5.9	6.1	42	2.9	1.7	1.7
15	19	12	12	43	2.9	2.3	2.2
16	13	10	9.0	44	3.1	2.0	1.6
17	13	8.0	9.0	45	2.7	1.9	1.5
18	7.0	6.0	5.0	46	1.9	1.7	2.2
19	7.4	4.7	4.6	47	3.0	1.7	1.8
20	5.2	3.9	4.3	48	4.0	2.7	3.1
21	11	4.6	5.5	49	2.0	1.5	1.6
22	5.3	3.5	2.6	50	3.3	1.9	1.9
23	6.6	5.0	6.5	51	2.7	1.7	1.6
24	4.5	3.0	3.4	52	2.8	1.8	1.8
25	4.4	3.9	3.1				
26	4.5	3.2	4.2				
27	5.1	3.8	4.0				
28	7.0	3.7	3.9	Average	5.8	4.0	4.1

Table 4.1.3 Radioparticulate Fallout—1975
(Gummed Paper Data—Station Weekly Average)

Station Number	Location	Long-Lived Beta Activity 10^{-4} $\mu\text{Ci}/\text{ft}^2$	Total Particles Per Sq. Ft.*
<u>Laboratory Area</u>			
HP-1	S 3587	0.14	0.58
HP-2	NE 3025	0.10	0.0
HP-3	SW 1000	0.09	0.0
HP-4	W Settling Basin	0.12	0.0
HP-5	E 2506	0.10	0.0
HP-6	SW 3027	0.08	0.19
HP-7	W 7001	0.06	0.0
HP-8	Rock Quarry	0.07	13
HP-9	N Bethel Valley Road	0.08	0.38
HP-10	W 2075	0.09	0.77
HP-16	E 4500	0.09	0.38
HP-20	HFIR	0.07	0.0
Average		0.09	1.3
<u>Perimeter Area</u>			
HP-31	Kerr Hollow Gate	0.09	
HP-32	Midway Gate	0.07	
HP-33	Gallaher Gate	0.07	
HP-34	White Oak Dam	0.08	
HP-35	Blair Gate	0.08	
HP-36	Turnpike Gate	0.08	
HP-37	Hickory Creek Bend	0.07	
HP-38	E EGCR	0.09	
HP-39	Townsite	0.10	
Average		0.08	
<u>Remote Area</u>			
HP-51	Norris Dam	0.07	
HP-52	Loudoun Dam	0.06	
HP-53	Douglas Dam	0.06	
HP-54	Cherokee Dam	0.07	
HP-55	Watts Bar Dam	0.07	
HP-56	Great Falls Dam	0.07	
HP-57	Dale Hollow Dam	0.08	
HP-58	Knoxville	0.06	
Average		0.07	

*Data determined from autoradiograms. Gummed paper from perimeter and remote areas were not processed by this method during 1975.

Table 4.1.4 Concentration of Beta Radioactivity in Rainwater—1975
(Weekly Average by Stations)

Station Number	Location	Activity in Collected Rainwater, 10^{-8} $\mu\text{Ci/ml}$
<u>Laboratory Area</u>		
HP-7	West 7001	1.6
<u>Perimeter Area</u>		
HP-31	Kerr Hollow Gate	1.5
HP-32	Midway Gate	1.2
HP-33	Gallaher Gate	1.6
HP-34	White Oak Dam	1.7
HP-35	Blair Gate	1.8
HP-36	Turnpike Gate	1.6
HP-37	Hickory Creek Bend	1.0
HP-38	E EGCR	2.2
HP-39	Townsite	1.4
Average		1.5
<u>Remote Area</u>		
HP-51	Norris Dam	2.5
HP-52	Loudoun Dam	2.7
HP-53	Douglas Dam	2.8
HP-54	Cherokee Dam	3.0
HP-55	Watts Bar Dam	1.6
HP-56	Great Falls Dam	2.7
HP-57	Dale Hollow Dam	2.6
HP-58	Knoxville	1.5
Average		2.4

Table 4.1.5 Weekly Average Concentration of Beta
Radioactivity in Rainwater—1975
(Units of 10^{-8} $\mu\text{Ci/ml}$)

Week Number	LAM's	PAM's	RAM's	Week Number	LAM's	PAM's	RAM's
1	2.8	2.7	3.8	29	0.90	2.1	3.0
2	3.2	2.0	2.7	30	2.9	2.2	2.4
3	3.2	3.3	3.7	31	*	1.1	2.6
4	1.2	1.2	2.1	32	2.1	0.67	1.5
5	2.7	2.7	3.0	33	*	0.51	1.2
6	4.1	4.5	4.2	34	*	*	1.9
7	3.2	4.8	7.0	35	*	*	0.6
8	2.2	2.0	3.1	36	0.10	*	0.50
9	3.2	1.4	4.6	37	*	0.23	0.72
10	3.2	3.9	5.3	38	0.60	0.19	0.71
11	3.5	3.7	4.6	39	*	0.16	0.32
12	2.6	2.8	4.6	40	0.20	0.37	0.64
13	0.9	0.2	0.3	41	1.3	0.47	1.1
14	2.5	2.5	4.0	42	0.50	0.76	1.1
15	2.0	1.9	4.0	43	0.20	0.74	0.52
16	2.1	2.5	3.0	44	*	*	0.57
17	1.6	2.0	2.5	45	*	0.57	2.0
18	*	*	3.2	46	0.50	0.67	1.5
19	2.6	2.3	3.3	47	*	0.16	1.1
20	1.1	1.3	2.2	48	0.20	1.2	1.1
21	*	*	2.9	49	0.10	0.41	2.1
22	*	0.88	2.3	50	1.1	0.52	2.0
23	0.60	0.91	0.95	51	1.3	*	1.1
24	1.6	1.1	1.5	52	0.9	1.0	1.5
25	*	*	4.5				
26	2.1	1.3	3.3				
27	*	1.7	3.6				
28	*	*	4.6	Average	1.6	1.5	2.4

*No rainfall.

Table 4.1.6 Weekly Concentration of ^{131}I in Air—1975
(Units of 10^{-14} $\mu\text{Ci/cc}$)

Week Number	LAM's	PAM's	Week Number	LAM's	PAM's
1	1.9	0.6	29	1.5	0.7
2	2.6	0.8	30	1.4	0.5
3	2.3	0.6	31	2.6	0.8
4	3.5	0.4	32	1.7	0.8
5	2.5	0.6	33	2.8	0.4
6	2.3	0.6	34	2.9	0.5
7	5.0	0.6	35	2.2	0.8
8	3.4	0.6	36	1.4	0.5
9	3.3	0.9	37	2.3	0.7
10	1.5	0.5	38	0.9	0.5
11	1.5	0.4	39	1.1	0.4
12	0.9	0.6	40	2.5	0.9
13	2.1	0.8	41	1.6	0.5
14	3.1	0.8	42	1.2	0.6
15	3.1	0.8	43	2.0	0.5
16	3.1	0.9	44	2.6	0.8
17	2.0	0.6	45	2.3	0.8
18	2.1	0.8	46	3.4	0.9
19	1.9	0.6	47	2.7	0.5
20	2.2	0.8	48	2.4	1.1
21	6.9	0.9	49	1.5	0.3
22	3.6	0.9	50	1.4	0.6
23	2.4	0.9	51	1.8	1.1
24	4.1	0.7	52	2.5	0.7
25	6.5	0.6			
26	2.8	0.7			
27	2.0	0.4			
28	2.4	0.4	Average	2.5	0.7

Table 4.1.7 Concentration of ^{131}I in Air—1975
(Weekly Average by Stations)

Station Number	Location	Activity in Air 10^{-14} $\mu\text{Ci/cc}$
<u>Laboratory Area</u>		
HP-4	W Settling Basin	2.9
HP-6	SW 3027	3.3
HP-7	W 7001	2.6
HP-8	Rock Quarry	1.7
HP-9	N Bethel Valley Road	2.2
HP-10	W 2075	3.2
HP-16	E 4500	2.5
HP-20	HFIR	1.7
Average		2.5
<u>Perimeter Area</u>		
HP-31	Kerr Hollow Gate	0.7
HP-32	Midway Gate	0.8
HP-33	Gallagher Gate	0.6
HP-34	White Oak Dam	0.8
HP-35	Blair Gate	0.7
HP-36	Turnpike Gate	0.6
HP-37	Hickory Creek Bend	0.6
HP-38	E EGCR	0.6
HP-39	Townsite	0.6
Average		0.7

Table 4.1.8 Specific Radionuclides in Air—1975
(Composite Samples)
10⁻¹⁵ µCi/ml

Quarter	⁷ Be	⁵⁴ Mn	⁶⁰ Co	⁹⁰ Sr	⁹⁵ Zr- ⁹⁵ Nb	¹⁰⁶ Ru	¹²⁵ Sb	¹³⁷ Cs	¹⁴⁴ Ce	²²⁸ Th	²³⁰ Th	²³² Th	²³⁴⁻²³⁵ U	²³⁸ U	²³⁹ Pu	²⁴⁰ Pu
1	160	< 0.32	< 0.32	ND	22	16	1.8	3.0	34							
2	170	< 0.32	< 0.15	ND	54	13	1.7	3.2	20							
3	148	ND**	< 0.24	ND	0.78	< 3.8	ND	3.0	2.4							
4	176	ND	ND	ND	≤ 0.21	< 3.2	ND	0.41	1.5							
Laboratory Area*																
Perimeter Area																
1	93	0.20	ND	0.52	37	10	1.6	1.8	17	0.019	0.021	0.016	0.84	0.35	0.00031	0.018
2	130	0.31	ND	0.92	9.8	12	2.1	2.9	25	0.014	0.032	0.013	1.2	0.36	0.0017	0.030
3	100	ND	ND	0.55	3.4	< 3.0	< 0.37	0.67	3.9	0.014	0.026	0.018	1.1	0.34	< 0.00046	0.0057
4	100	ND	ND	1.0	ND	< 2.1	ND	< 0.41	1.1	0.0081	0.018	0.0086	1.4	0.65	< 0.00046	0.0021
Remote Area																
1	120	0.23	ND	0.51	58	12	1.6	2.0	21	0.014	0.016	0.013	0.076	0.044	0.00087	0.018
2	160	0.29	ND	0.76	23	16	2.0	3.0	27	0.015	0.020	0.013	0.057	0.021	0.00068	0.028
3	100	ND	ND	0.50	< 1.2	< 4.1	ND	0.81	5.1	0.015	0.014	0.0099	0.091	0.039	< 0.00046	0.0073
4	110	ND	ND	0.15	ND	1.7	ND	0.35	0.93	0.0031	0.0065	0.0036	0.083	0.044	< 0.00046	0.0021

* Wet chem. analysis on local filters not complete at time of report.

**None Detected.

Table 4.1.9 Concentration of ^{131}I in Raw Milk—1975

Station Number	No. of Samples	Units of 10^{-9} $\mu\text{Ci/ml}$			Comparison with Standard ^b
		Maximum	Minimum ^a	Average	
<u>Immediate Environs^c</u>					
1	46	5.4	< 0.45	< 0.68	FRC Range 1
2	47	5.0	"	< 0.59	"
3	"	2.7	"	< 0.56	"
4	33	1.3	"	< 0.49	"
5	47	2.5	"	< 0.53	"
6	"	1.6	"	< 0.50	"
7	"	1.2	"	< 0.51	"
Average				0.56 \pm 0.03	"
<u>Remote Environs^d</u>					
51	9	0.45	< 0.45	< 0.45	FRC Range 1
52	10	1.0	"	< 0.52	"
53	7	0.45	"	< 0.45	"
54	10	0.45	"	< 0.45	"
Average				0.47 \pm 0.05	"

^aMinimum detectable concentration of ^{131}I is 0.45×10^{-9} $\mu\text{Ci}/\text{ml}$.

^bApplicable FRC standard, assuming 1 liter per day intake:

- | | | |
|-----------|--|---|
| Range I | 0 to 1×10^{-8} $\mu\text{Ci}/\text{ml}$ | - Adequate surveillance required to confirm calculated intakes. |
| Range II | 1×10^{-8} $\mu\text{Ci}/\text{ml}$ to 1×10^{-7} $\mu\text{Ci}/\text{ml}$ | - Active surveillance required. |
| Range III | 1×10^{-7} $\mu\text{Ci}/\text{ml}$ to 1×10^{-6} $\mu\text{Ci}/\text{ml}$ | - Positive control action required. |

Note: Upper limit of Range II can be considered the concentration guide.

^cSee Figure 4.0.6.

^dSee Figure 4.0.7.

Table 4.1.10 Concentration of ^{90}Sr in Raw Milk—1975

Station Number	No. of Samples	Units of 10 ⁻⁹ μCi/ml			Comparison with Standard ^b
		Maximum	Minimum ^a	Average	
<u>Immediate Environs^c</u>					
1	48	4.6	1.7	3.2	FRC Range 1
2	"	3.9	1.0	2.5	
3	"	5.1	1.3	3.3	"
4	35	4.0	1.0	2.2	"
5	48	7.3	3.0	5.0	"
6	"	10	3.1	6.1	"
7	34	3.9	1.2	2.7	"
Average				3.7 ± 0.10	"
<u>Remote Environs^d</u>					
51	9	3.9	2.4	3.1	FRC Range 1
52	10	7.2	0.70	2.2	"
53	7	4.6	2.8	3.5	"
54	9	4.7	3.1	4.1	"
Average				3.2 ± 0.37	"

^aMinimum detectable concentration of ^{90}Sr in milk is 0.5×10^{-9} $\mu\text{Ci/ml}$.

^bApplicable FRC Standard, assuming 1 liter per day intake:

- | | | |
|-----------|--|---|
| Range I | 0 to 2×10^{-8} $\mu\text{Ci/ml}$ | - Adequate surveillance required to confirm calculated intakes. |
| Range II | 2×10^{-8} $\mu\text{Ci/ml}$ to 2×10^{-7} $\mu\text{Ci/ml}$ | - Active surveillance required. |
| Range III | 2×10^{-7} $\mu\text{Ci/ml}$ to 2×10^{-6} $\mu\text{Ci/ml}$ | - Positive control action required. |

Note: Upper limit of Range II can be considered the concentration guide.

^cSee Figure 4.0.6.

^dSee Figure 4.0.7.

Table 4.1.11 Discharge of ^{131}I from ORNL Stacks—1975^a

Stack Number ^b	Curies	
	Total for Year	Monthly Average
3039	1.9	0.16
7911	0.15	0.01
Total	2.1	0.17

^aData furnished by Operations Division.

^bNo detectable levels of ^{131}I were discharged from Stack Numbers 2026, 3020, and 7512.

Table 4.2.1 Annual Discharges of Radionuclides to the Clinch River
(Curies)

Year	^{137}Cs	^{106}Ru	^{90}Sr	^{95}Zr	^{95}Nb	Trans U Alpha	^3H
1968	1.1	5.2	2.8	0.27	0.27	0.04	9700
1969	1.4	1.7	3.1	0.18	0.18	0.2	12200
1970	2.0	1.2	3.9	0.02	0.02	0.4	9500
1971	0.93	0.50	3.4	0.01	0.01	0.05	8900
1972	1.7	0.52	6.5	0.01	0.01	0.05	10600
1973	2.3	0.69	6.7	0.05	0.05	0.08	15000
1974	1.2	0.22	6.0	0.02	0.02	0.02	8600
1975	0.62	0.30	7.2	NA*	NA	0.02**	11000
7.5 Σ 8.14							

*NA - No analysis performed.

**Radionuclides identified from yearly composite sample. Activity composed of ^{238}Pu , 19%; ^{239}Pu , 14%; ^{244}Cm , 49%; and ^{241}Am , 18%.

Table 4.2.2 Radionuclides in the Clinch River—1975

Location	No. of Samples	Range	Concentration of Radionuclides of Primary Concern Units of 10^{-9} $\mu\text{Ci/ml}$				% MPC ^c
			⁹⁰ Sr	¹³⁷ Cs	¹⁰⁶ Ru	³ H	
C-2 CRM 23.1 ^a	4	Max.	0.14	0.05	0.09	820	
		Min.	0.05	0	0.09	630	
		Avg.	0.07 \pm 0.02	0.02 \pm 0.01	0.09	730 \pm 52	0.05
CRM 20.8 ^b	12	Max.	2.42	0.17	0.09	6000	
		Min.	0.40	0.03	0.01	170	
		Avg.	1.20 \pm 0.19	0.09 \pm 0.01	0.05 \pm 0.01	1800 \pm 450	0.49
C-3 CRM 14.5 ^a	4	Max.	1.32	0.14	0.18	4100	
		Min.	0.09	0.05	0.09	2000	
		Avg.	0.42 \pm 0.30	0.07 \pm 0.02	0.13 \pm 0.02	2500 \pm 530	0.23
C-5 CRM 4.5 ^a	4	Max.	0.82	0.05	0.27	1400	
		Min.	0.05	0.05	0.09	600	
		Avg.	0.31 \pm 0.18	0.05	0.19 \pm 0.05	1100 \pm 190	0.15

^aMeasured values in the Clinch River.^bValues given for this location are calculated values based on the concentrations measured at White Oak Dam (Station W-1) and the dilution afforded by the Clinch River. They do not include radioactive materials (e.g., fallout) that may enter the river upstream of White Oak Creek outfall (CRM 20.8).^cMost restrictive concentration guide for each isotope used for calculating percent concentration guide. The method for calculating percent of concentration guide for a known mixture of radionuclides is given in ERDA Manual, Appendix 0524, Annex A.

Table 4.2.3 Calculated Percent MPC_w
of ORNL Radioactivity Releases in Clinch River Water
Below the Mouth of White Oak Creek—1975

Month	% MPC _w
January	0.78
February	0.32
March	0.78
April	0.24
May	0.39
June	0.40
July	0.15
August	0.15
September	0.40
October	0.91
November	0.77
December	0.54
Average	0.49

Table 4.2.5 Annual Average Percent MPC_w
of Tritium in the Clinch River

Year	CRM 20.8 ^a
1968	0.07
1969	0.11
1970	0.05
1971	0.04
1972	0.04
1973	0.07
1974	0.04
1975	0.06

^aValues given are calculated from the level of waste released from White Oak Dam and dilution provided by the Clinch River.

Table 4.2.6 Radionuclide Content of Clinch River Fish—1975

Species	No. of Samples	pCi/kg Wet Weight						Estimated % MBIB
		⁹⁰ Sr	⁶⁰ Co	¹⁰⁶ Ru	¹³⁷ Cs	^{110m} Ag	¹²⁵ Sb	
Clinch River Mile 41.0 (Above ORNL Waste Outfall)								
White Crappie	1	140	41	130	490	72	28	1.0
Carp	1	46	5.9	25	29	20	10	0.33
Clinch River Mile 14.5 (Below ORNL Waste Outfall)								
White Crappie	1	220	45	230	30	90	39	1.6
Carp	1	13	14	26	17			0.10

^aComposite of ten fish in each species.

^bMaximum Permissible Intake - Intake of radionuclides from eating fish is calculated to be equal to a daily intake of 2.2 liters of water, over a period of one year, containing the concentration guide of the radionuclides in question. Consumption of fish is assumed to be 37 lb/yr of the species in question.

Table 4.2.4 Annual Average Percent MPC_w of Beta Emitters,
Other than Tritium, in the Clinch River

Year	CRM 23.1 ^a	CRM 20.8 ^b	CRM 14.5 ^a	CRM 4.5 ^a
1968	0.17	0.83	0.37	0.52
1969	0.30	0.36	0.48	0.41
1970	0.22	0.27	0.53	0.47
1971	0.21	0.20	0.65	0.44
1972	0.18	0.26	0.58	0.48
1973	0.24	0.49	0.47	0.62
1974	0.06	0.36	0.26	0.21
1975	0.03	0.43	0.14	0.12

^aValues given for this location are based on analyses of water taken directly from the river.

^bValues given for this location are calculated from the levels of radionuclides released from White Oak Dam and dilution provided by the Clinch River. The contribution from upstream as measured at CRM 23.1 is not included.

Table 4.3.1 External Gamma Radiation Measurements—1975

Station Number	Location	Background	
		$\mu\text{R/hr}$	mrem/yr ^a
<u>Perimeter Stations^b</u>			
HP-31	Kerr Hollow Gate	9.1	80
HP-32	Midway Gate	10	88
HP-33	Gallaher Gate	8.9	78
HP-34	White Oak Dam	14	123
HP-35	Blair Gate	7.6	67
HP-36	Turnpike Gate	7.4	65
HP-37	Hickory Creek Bend	7.6	67
HP-38	E EGCR	7.7	67
HP-40	Melton Hill	5.7	50
Average		8.7	76
<u>Remote Stations^c</u>			
HP-51	Norris Dam	5.8	51
HP-52	Loudoun Dam	6.9	60
HP-53	Douglas Dam	7.0	61
HP-54	Cherokee Dam	6.7	59
HP-55	Watts Bar Dam	6.5	57
HP-56	Great Falls Dam	6.3	55
HP-57	Dale Hollow Dam	8.1	71
HP-58	Knoxville	11	96
Average		7.3	64

^aCalculated assuming that an individual remained at this point for 24 hours/day for the entire year.

^bSee Figure 4.0.3.

^cSee Figure 4.0.4.

Table 4.3.2 External Gamma Radiation Measurements Along the
Perimeter of the ERDA-Oak Ridge Controlled Area—1975

Location ^a	$\mu\text{R/hr}$	mrem/yr ^b
HP-41	17	150
HP-42	25	220
HP-60	14	120
HP-61	20	180
HP-62	36	320
HP-63	72	630
HP-64	32	280
HP-65	40	350
HP-66	43	380
HP-67	26	230
HP-68	14	120
HP-69	11	100
Average	29	260

^aSee Figure 4.0.8.

^bCalculated assuming that an individual remained at this point for 24 hours/day for the entire year.

Table 4.4.1 Soil Samples from Near Perimeter
Air Monitoring Stations—1975

Sampling ^a Location	No. of Samples ^b	Dry Soil ^c Units of 10^{-8} $\mu\text{Ci/g}$	
		Plutonium(α)	Uranium(α) ^d
HP-31	1	3.3	180
HP-32	1	5.4	170
HP-33	1	3.7	29
HP-34	1	5.0	33
HP-35	1	5.0	29
HP-36	1	3.6	35
HP-37	1	2.1	26
HP-38	1	2.5	35
HP-39	1	4.3	59
Average		3.9	66

^aSee Figure 4.0.3.

^bNine samples, approximately three inches in diameter and one centimeter thick, collected in a one-square-meter area at each location and composited for analysis.

^cApplicable guides for soil contamination have not been established.

^dBased on "special" uranium curie. NCRP - Specification of Units for Natural Uranium and Natural Thorium.

Table 4.5.1 Environmental Monitoring Samples—1975

Sample Type	Type of Analyses	Number of Samples
Monitoring Network Air Filters	Gross Alpha, Gross Beta	1664
Monitoring Network Air Filters	Autoradiogram	1248
Monitoring Network Air Filters	Gamma Spectrometry, Wet-Chemistry	12 Groups
Gummed Paper Fallout Trays	Autoradiogram	1248
Gummed Paper Fallout Trays	Long-Lived Activity Count	1664
Charcoal Cartridge	^{131}I	1248
Fish	Radiochemical, Gamma Spectrometry	8
Rainwater	Gross Beta	938
Raw Milk	^{131}I , ^{90}Sr	468
White Oak Dam Effluent	Gross Beta, Radiochemical, Gamma Spectrometry	408
White Oak Creek	Gross Beta, Radiochemical, Gamma Spectrometry	236
Clinch River Water	Radiochemical, Gamma Spectrometry	54
Potable Water	Radiochemical, Gamma Spectrometry	20
Soil Samples	Plutonium and Uranium	18

5.0 RADIATION AND SAFETY SURVEYS

5.1 Laboratory Operations Monitoring

During 1975, Radiation and Safety Surveys personnel assisted the operating groups in keeping the contamination, air concentration, and personnel exposure levels well below the established maximum permissible limits. Through seminars, safety meetings, and informal discussions with supervision, they assisted in reducing or eliminating a number of problems associated with radiation protection at the Laboratory. The following is a brief description of some of the activities monitored during the year.

5.1.1 Bulk Shielding Reactor, Building 3010

Monitoring and surveillance were provided for the operations associated with the repair of leaks in the walls and floor of the Bulk Shielding Reactor pool. This was essentially the same operation as the one conducted in 1974 except that in the 1975 operation all fuel elements along with the 2.3×10^4 Ci ^{60}Co source were transferred to the Graphite Reactor canal for storage. As in 1974, other equipment such as D_2O tanks, reactor vessel, and grid assembly were moved north to the deep pit area and submerged in approximately eight feet of water. Radiation levels from this equipment presented a minor problem in that working time was restricted at two locations. The work was completed with personnel exposures totaling only a small percentage of the permissible level. Surface contamination did not present a significant problem.

5.1.2 Radiochemical Pilot Plant Operations, Building 3019

Radiation and Safety Surveys personnel continued to provide monitoring and surveillance for the Chemical Technology Division's operations in support of the Light Water Breeder Reactor program. During the year, ~ 500 kg of ^{233}U and $\sim 10^4$ kg ^{232}Th were recovered through dissolution operations. Solvent Extraction and Ion Exchange Purification Systems processed ~ 522 kg and ~ 810 kg of ^{233}U , respectively. The Oxide Conversion Facility produced ~ 757 kg of ceramic grade $^{233}\text{UO}_2$. About 684 kg of $^{233}\text{UO}_2$ were packaged and shipped off-site.

Two hundred forty-seven Radiation Work Permits were certified for the more hazardous operations. These included: replacement of two cracked alpha containment box windows, Room 502; clean out of partially plugged hydrogen furnace exhaust venturis, Room 502; replacement of the 3020 stack steam fan and associated ducts; replacement of process off-gas filters and filter housing, Building 3121; and direct maintenance and installation of process equipment in grossly contaminated "hot" cells.

No significant internal exposure resulted from the above operations. There were a few minor releases of radioactive material which were confined to established zones where control measures were effective. The maximum annual whole body dose equivalent to personnel from external sources (principally β, γ from ^{232}U daughters) was 2.71 rem.

5.1.3 Decontamination of Hot Cell #2, Building 3029

This cell had not been opened since 1970 and was contaminated with ^{137}Cs and ^{147}Pm . A plastic tent was built outside the cell door to serve as a change room. Readings ranged from 800 mrad/hr at the cell door to 15 rad/hr at various points in the cell. Decontamination was done by Operations Division personnel and remodeling inside the cell was done by Plant and Equipment Division personnel. Personnel exposure controls were effective, and there was no spread of contamination.

5.1.4 Oak Ridge Research Reactor 24-Inch Water Line, Building 3085

Surveillance and assistance were provided during excavation and repair of leaks in the 24-inch primary water line of the Oak Ridge Research Reactor. The excavation area extended from #1 pump cell to approximately 80 feet north of Building 3085 to the "Y" pit. Radiation levels were encountered to 2 R/hr and transferable contamination, which resulted from the leaks, was found up to 100 mR/hr. The contamination was primarily ^{115}Cd with traces of ^{24}Na , ^{46}Sc , ^{51}Cr , ^{58}Co , ^{60}Co , ^{95}Zr , ^{137}Cs and ^{141}Ce . All contaminated soil was transferred to Solid Waste Disposal Area #6.

Following repair of the leaks, a six-inch concrete wall was poured on each side of the pipe and covered with 3/8-inch aluminum treadplate.

Exposure to personnel involved in the operation was well within permissible limits.

5.1.5 Cleanup of Building 3503

A major accomplishment involving Radiation and Safety Surveys personnel was the cleanup of Building 3503. Starting in July, 1974, all contaminated equipment located outside the cells was removed; the building was decontaminated and painted from the roof to the floor; and the floor was then capped with two inches of new concrete which was painted with epoxy. Except for the cells, the "C" Zone restrictions were removed from the building in November, 1975.

Somewhat less effort, involving the same technique and operation, resulted in the removal of the "C" Zone restrictions from Building 2528 in August, 1975. The filter plenum is the only remaining "C" Zone within this building.

The vacuum system serving the five laboratories in Building 3508 began leaking around midyear. Examination resulted in condemnation. The vacuum tank, filter box, and all piping were removed during the last quarter of 1975. Working inside a plastic tent with the pipe and saws enclosed in a plastic sleeve, the workmen wrapped each section in heavy plastic and removed them from the tent to a plywood box. Close cooperation of Chemical Technology Division, Plant and Equipment Division, and Radiation and Safety Surveys personnel resulted in the removal of the entire system, containing several millicuries of alpha contaminants, with no serious personnel contamination and only a minor spread of activity outside the tent.

5.1.6 Hot Cells, Building 4501

Planned experimental work using irradiated materials required the upgrading of the existing ventilation system from the four cells in Building 4501. Modifications were made to increase exhaust capacity and add filtration for individual cells. Radiation and Safety Surveys surveillance was provided for the disconnection and removal of the old contaminated plenum and ductwork and the installation of new booster exhaust fans, ductwork, and high efficiency filtration. Radiological control procedures limited the release of radioactivity from inside the highly contaminated (15,000 α d/m and 40,000-50,000 β, γ d/m) ductwork to insignificant levels. No personnel internal exposures occurred and external exposures were minimal.

5.1.7 High Level Chemical Development Facility, Building 4507

The planned LMFBR and LWR studies in Building 4507 required the complete renovation of the Curium Recovery Facility. This facility consisted of two inter-connected hot cells and a number of glove boxes, pumping stations, and control and operating areas spread over three floors.

In late February, 1975, with continuous Radiation and Safety Surveys surveillance, the removal of all equipment, structural steel and piping in preparation for a complete decontamination of the facility was begun. The glove boxes, cells, piping and equipment were grossly contaminated with ^{244}Cm , ^{241}Am , ^{238}Pu and mixed fission products. After extensive remote washing, a health physics survey indicated the general radiation background in the cells to be about 500 mrem/hr with spots as high as 10 rem/hr. Wipe tests indicated transferable alpha contamination in excess of 2.5×10^6 d/m.

ORNL craftsmen (pipefitters, welders, and millwrights) entered the cells dressed in positive air supply plastic suits and cut the piping (Hastelloy, stainless steel, Zircaloy and tantalum), structural steel and mechanical equipment into short lengths by "hot rodding" with an electric welding machine. Chemical Technology Division personnel removed the scrap from the cells and packaged it for disposal at the Solid Waste Storage Area.

Cooperation between all individuals in the careful planning and execution of the project has been excellent. With the exception of two instances when plastic suits were accidentally punctured, no significant personnel contamination was detected. The results of a surveillance program for whole body counting and bio-assay of excreta samples indicated no significant internal exposure. External exposures were kept well below the permissible limits.

As of December 31, 1975, the removal of piping, equipment, etc., was about 90 percent complete. Decontamination of the equipment (glove) box and cells is expected to begin in late January, 1976.

5.1.8 Tower Shielding Facility, Building 7702

Some special radiation survey assistance was provided during initial operations of the TSR-II Research Reactor in the new fixed shield installation with large beam hole capability in the east shield face. Radiation measurements made in the beam line (no shielding) at ~ 1300 feet showed ~ 23 mrem/hr when the reactor was operating at 5 kW. With the beam unshielded and the reactor at 1 kW, the radiation level at the access control gate in the 600-foot fence was ~ 4.5 mrem/hr.

5.1.9 High Flux Isotope Reactor (HFIR), Building 7900

Extensive Radiation and Safety Surveys assistance was provided during a three-month shutdown of the reactor for the replacement of the permanent outer ring of the beryllium reflector and other repair and maintenance operations. The potential for high exposure doses and for the release of large amounts of contamination was always present during these operations, most of which had not been performed previously. Detailed planning and precise execution using specially designed remote tools were required for the new and difficult procedures.

The major operations of the program involved removal of most of the components in the reactor pressure vessel, removal of the four slant (EF) beam tubes, and the four horizontal beam tubes along with their associated 45-ton spectrometer shields. Major maintenance was performed on the control plate drive system and the primary heat exchanger systems, both of which were contaminated with ^{60}Co and ^{244}Cm . Scale deposits were removed from the heat exchangers by a special team from the Dow Chemical Company.

Air activity levels did not exceed the normal background during the shutdown, and exposure doses to the people involved in the program were kept well below the quarterly limit. Transferable contamination was confined to the room in which the various operations were performed, and no extensive decontamination efforts were necessary.

5.1.10 Transuranium Processing Plant (TRU), Building 7920

Close surveillance by Radiation and Safety Surveys personnel was provided during the hot cells repair and modification program and also during the extensive handling of waste while the building was processing HFIR target rods. Californium-252 sources of up to 50 mg were fabricated and transferred, irradiated target rods were unloaded from carriers into a cell, and high level alpha and neutron emitting samples were handled and analyzed. Surveillance and monitoring by Radiation and Safety Surveys personnel were also provided for these operations.

5.1.11 Thorium-Uranium Recycle Facility (TURF), Building 7930

Waste contaminated with ^{252}Cf was removed from Cell G and transferred to the burial ground in six drums. The drums were contained in plastic and sealed in concrete casks having walls four inches in thickness. Dose

rates through the cask walls were to 400 mrem/hr gamma and 650 mrem/hr neutrons. No airborne or transferable contamination was released, and exposure doses were minimal.

Thoughtful planning and very careful execution was required to prevent the release of alpha contamination during this operation. Radiation and Safety Surveys personnel aided in the planning and provided surveillance.

5.1.12 Cyclotron Production Operations, Building 9201-2

Through the cooperative effort of the ORNL Cyclotron Group and ORAU, a very promising cancer detecting isotope, ^{11}C , is being produced for research in the 86" Cyclotron.

These groups had previously researched the production and use of ^{67}Ga for locating and imaging tumors. Gallium is now being used worldwide. The Cyclotron continues to serve as backup supplier to the commercial cyclotrons.

5.1.13 Shale Fracturing Injections

Three radioactive liquid waste injections were made at the Shale Fracturing facility during the year which necessitated full-time Radiation and Safety Surveys monitoring during these periods. The Halliburton Company personnel, under contract with ORNL, made the injections. All personnel exposures were kept below maximum permissible levels, and personnel contamination control was adequate. Also, in October, 1975, monitoring was provided for Halliburton personnel when 20 curies of ^{198}Au were injected into the new well for geological experimental purposes. Again, personnel exposures and contamination controls were adequate.

5.1.14 X-Ray Safety Program

In a continuing effort to update the safeguards on X-ray equipment, the following was done: a re-evaluation of the safety systems on 40 X-ray diffraction and X-ray fluorescence units was made. A program was proposed and initiated to bring the safety systems on the most hazardous facilities more in line with the recommendations of ANSI N43.2-1971 (Radiation Safety for X-Ray Diffraction and Fluorescence Analysis Equipment). One prototype safety system was installed; an improved safety system was installed at a walk-in X-ray facility that was being reactivated.

Two new walk-in X-ray facilities were put into service. Both of these units were reviewed by Instrumentation and Controls and Health Physics Division personnel for electrical safety, interlock system integrity and radiation safety.

The annual survey of X-ray machines was completed. The review focuses mainly on clear identification of work areas and X-ray machines, X-ray leakage measurements, interlock system and safety device testing, identity of persons responsible for operating X-ray machines, noting equipment or

experiment change and reviewing procedures. The survey revealed that:

- (1) A device used to detect a warning light failure was out of order.
- (2) An old fluoroscope was found to be back in service after being out of service for a number of years. The safety system was reviewed and updated to conform with the current safety requirements.
- (3) A warning light on an X-ray diffraction unit was too dim to be an obvious warning. The standard warning light system was recommended.
- (4) Warning signs had temporarily been removed at one facility because of walls in the area being painted.

5.1.15 Microwave Safety Program

Microwave ovens still constitute the major source of potential microwave hazard at ORNL. There are now 14 known microwave ovens. These ovens were surveyed quarterly, and microwave leakage was found to be below federal limits except in one case. The oven was not yet leaking at a dangerous level but was replaced with a more suitable one.

5.2 American Nuclear Corporation Clean-Up Project

This project began in October, 1974, and continued until May, 1975.¹ Radiation and Safety Surveys monitoring was provided on a full-time basis during this period. All personnel exposures were kept below maximum permissible levels and contamination controls were adequate.

5.3 Unusual Occurrences

Radiation incidents are classified according to a severity index system developed over the past several years (see ORNL-3665, Applied Health Physics Annual Report for 1963, pp. 14-15). The method serves to index unusual occurrences according to degree of severity and permits a system of analysis regarding Applied Health Physics and Safety practices among Laboratory operations.

During 1975, there were 11 unusual occurrences recorded which is one more than the number recorded for 1974 (Table 5.3.1, page 61). The frequency rate of unusual occurrences among Laboratory divisions involved (Table 5.3.2, page 62) is known to vary in relationship to the quantity of radioactive materials handled, the number of radiation workers involved, and the radiation potential associated with a particular operation or facility.

5.4 Laundry Monitoring

There were approximately 537,000 articles of wearing apparel monitored at the Laundry during 1975. Approximately 10 percent were found con-

¹See Applied Health Physics and Safety Annual Report for 1974, ORNL-5055, p. 44.

taminated. Of 393,213 khaki garments monitored during the year, only 158 were found contaminated.

A total of 5,539 full-face respirators and 4,547 cannisters were monitored during the year; and of this number, 996 required further decontamination after the first cleaning cycle.

Table 5.3.1 Unusual Occurrences Summarized for the 5-Year Period Ending with 1975

	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>1975</u>
Number of Unusual Occurrences Recorded . . .	10	11	10	10	11
A. Number of incidents of minor consequence involving personnel exposure below MPE limits and requiring little or no cleanup effort	1	2	7	5	6
B. Number of incidents involving personnel exposure above MPE limits and/or resulting in special cleanup effort as the result of contamina- tion	9	9	3	5	5

Table 5.3.2 Unusual Occurrence Frequency Rate within the Divisions
for the 5-Year Period Ending with 1975

Division	No. of Unusual Occurrences					5-Year Total	Percent Lab. Total (5-Year Period)
	1971	1972	1973	1974	1975		
Analytical Chemistry	1	1				2	3.8
Biology				1	1	2	3.8
Chemical Technology	1	1	4	3	3	12	23.1
Chemistry	1	1		1	1	4	7.7
Engineering				1		1	1.9
Environmental Sciences			1			1	1.9
Health Physics			1			1	1.9
Isotopes	4	5		2		11	21.3
Metals and Ceramics	1				1	2	3.8
Neutron Physics					1	1	1.9
Operations	2	2	3	1	3	11	21.3
Physics		1				1	1.9
Reactor Chemistry			1			1	1.9
Solid State	—	—	—	<u>1</u>	<u>1</u>	<u>2</u>	<u>3.8</u>
TOTALS	10	11	10	10	11	52	100.0

6.0 INDUSTRIAL SAFETY AND SPECIAL PROJECTS

The safety record for 1975 was one of the best in the history of the Laboratory. Only two Disabling Injuries were sustained during the year. In 1975, ORNL employees worked a continuous period of 217 days (4,543,462 hours) without a Disabling Injury. For this accomplishment, the Laboratory earned the highest awards of both the Union Carbide Corporation and the National Safety Council, the Award of Distinguished Safety Performance and the Award of Honor. In addition, ERDA's Award of Honor was earned for a reduction of greater than 30 percent in the injury incidence rate compared with the 1974 record.

6.1 Accident Analysis

The Disabling Injury frequency rate for 1975 was 0.27. This rate has been surpassed only once, in 1968 when the Laboratory incurred one Disabling Injury (frequency rate 0.13). As the new OSHA system has a much broader base for recording injuries and illnesses, it is not feasible to make a comparison of the serious injuries that have been reported in past years under the old UCC's (ANSI Z16.1) serious injury classification system with the new OSHA classification system adopted in 1975. However, basing the number of Recordable Injuries and Illnesses on 200,000 employee-hours worked, the Laboratory had a frequency rate of 2.25 which was second best in the Nuclear Division. The injury statistics for ORNL for the period 1960-1975 are shown in Table 6.1.1, page 65. The Disabling Injury history of ORNL for the past five years is shown in Table 6.1.2, page 66; and the Disabling Injury frequency rates since the inception of Carbide's contract as compared with NSC, ERDA, and UCC are shown in Table 6.1.3, page 67.

Nine ORNL divisions did not have a Recordable Injury or Illness in 1975. Injury statistics by divisions are shown in Table 6.1.4, page 68.

Disabling Injury accident-free periods for ORNL are shown in Table 6.1.5, page 69. During 1975, from April 8 through November 10, the Laboratory accumulated over 4.5 million man-hours without a Disabling Injury. These figures represent the most hours worked since 1968-69 when the Laboratory accumulated an accident-free period of over 8.5 million man-hours.

Table 6.1.6, Figure 6.1.1, and Table 6.1.7, pages 70, 71, and 72, present ORNL injury data according to type, part of body injured and nature of injury.

A tabulation of injuries for the four UCC-ND facilities is shown in Table 6.1.8, page 73. ORNL had the lowest frequency rate for Disabling Injuries, 0.27, and a frequency rate of 2.25 for Recordable Injuries and Illnesses which was the second best for Nuclear Division facilities.

Statistics on motor vehicle accidents, fires, and off-the-job Disabling Injuries are shown in Tables 6.1.9, 6.1.10, and 6.1.11, pages 74 and

75. The number of off-the-job Disabling Injuries for ORNL employees was lower than at the other three plants within the Nuclear Division. ORNL had a frequency rate of 2.33 compared with ORGDP's frequency rate of 7.8, Paducah's 7.49, and Y-12's rate of 6.93. The off-the-job Disabling Injury frequency rate for the Nuclear Division was 5.05.

6.2 Summary of Disabling Injuries

Following are summaries of the two Disabling Injuries experienced at ORNL in 1975:

Date of Injury - 4/7/75

A health physicist was working on a 12' high flat concrete roof. Needing an object from below, he started down an unsecured aluminum extension ladder leaning against the edge of the roof. When he was about halfway down, the ladder started slipping to the right along the roof edge. He tried unsuccessfully to jerk it back upright but finally kicked it away to the right so as not to fall on it. He struck the ground on his left side, sustaining fractures to the left wrist and heel. Time loss: 147 days.

Date of Injury - 11/11/75

A research staff member of the Physics Division placed a small pot which had contained cesium in a sink, poured it about one-third full of alcohol, and waited several minutes. Observing no chemical reaction, he shook the pot, poured out the alcohol, and then looked into the pot. At that instant an explosion with flame occurred, ejecting glass and chemicals into his face. He suffered facial burns and a perforated cornea. Time loss: 26 days.

6.3 Safety Awards

A new Safety Incentive Plan was approved for employees of the Union Carbide Nuclear Division effective January 1, 1975. The plan stipulates that employees will be rewarded for successful achievement of Disabling Injury-free performance for a specific award period. The award period is defined as thirty days of continued operation in an installation without a Disabling Injury. As an added incentive, the monetary award value is escalated on a periodic basis for consecutive injury-free periods. The annual award for ORNL during 1975 amounted to \$13.50 for each employee which they will receive in the form of a merchandise item selected from a list of items chosen by the installation awards committee.

Table 6.1.1 ORNL Injury Statistics (1960-1975)

Year	Disabling Injuries			Serious Injuries	
	Number	Frequency	Severity	Number	Frequency
1960	6	0.94	77	99	15.5
1961	10	1.55	576	80	12.4
1962	10	1.45	377	70	10.2
1963	11	1.55	172	58	8.2
1964	8	1.07	148	83	11.1
1965	18	2.34	366	97	12.6
1966	5	0.64	155	93	11.9
1967	4	0.50	266	89	11.1
1968	1	0.13	8	73	9.4
1969	2	0.27	9	37	4.9
1970	5	0.76	88	49	7.5
1971	4	0.61	298	38	5.8
1972	7	1.08	52	49	7.6
1973	2	0.33	24	35	5.8
1974	5	0.81	51	30	4.9
1975	2	0.27	24	82	2.25*

*Based on the new OSHA system for recording injuries and illnesses (RII).

Table 6.1.2 Disabling Injury History—ORNL (1971-1975)

	1971	1972	1973	1974	1975
Number of Injuries	4	7	2	5	2
Labor Hours (Millions)	6.5	6.5	6.0	6.2	7.3
Frequency Rate	0.61	1.08	0.33	0.81	0.27
Days Lost or Charged	1944	337	692	315	173
Severity Rate	298	52	115	51	24

Table 6.1.3 ORNL Disabling Injury Frequency Rates Since Inception of Carbide Contract Compared with Frequency Rates for NSC, ERDA and UCC

Year	ORNL	NSC	ERDA	UCC
1948	2.42	11.49	5.25	5.52
1949	1.54	10.14	5.35	4.91
1950	1.56	9.30	4.70	4.57
1951	2.09	9.06	3.75	4.61
1952	1.39	8.40	2.70	4.37
1953	1.43	7.44	3.20	3.61
1954	0.79	7.22	2.75	3.02
1955	0.59	6.96	2.10	2.60
1956	0.55	6.38	2.70	2.27
1957	1.05	6.27	1.95	2.41
1958	1.00	6.17	2.20	2.21
1959	1.44	6.47	2.15	2.16
1960	0.94	6.04	1.80	1.92
1961	1.55	5.99	2.05	2.03
1962	1.45	6.19	2.00	2.28
1963	1.55	6.12	1.60	2.10
1964	1.07	6.45	2.05	2.20
1965	2.34	6.53	1.80	2.40
1966	0.64	6.91	1.75	2.57
1967	0.50	7.22	1.55	2.06
1968	0.13	7.35	1.27	2.24
1969	0.27	8.08	1.52	2.49
1970	0.76	8.87	1.28	2.27
1971	0.61	9.37	1.44	2.05
1972	1.08	10.17	1.40	1.73
1973	0.33	10.55	1.45	1.50
1974	0.81	10.20	1.60	0.99
1975	0.27	—	—	0.61

Table 6.1.4 Injury Statistics by Division—1975

Division	Medical Reports Received	Recordable Injuries and Illnesses		Disabling Injuries			Exposure Hours (In Millions)
		No.	Freq.	Number	Freq.	Sev.	
Analytical Chemistry	8	1	0.89				.224
Chemical Technology	25	8	3.16				.506
Chemistry	5	0	0				.181
Director's	2	0	0				.101
Physics	4	2	2.14	1	5.35	101	.187
Instr. and Controls	29	1	0.38				.530
Health Physics	8	1	0.68	1	3.42	503	.292
Metals and Ceramics	22	2	0.75				.531
Neutron Physics	1	0	0				.142
Computer Sciences	8	1	0.51				.390
Solid State	2	1	1.47				.136
Engineering	6	0	0				.440
Health	0	0	0				.061
Inspection Engineering	3	1	2.99				.067
Laboratory Protection	6	0	0				.179
Operations	49	6	2.84				.422
Employee Relations	10	0	0				.170
Plant and Equipment	295	51	6.10				1.671
Information	8	2	1.02				.391
Isotopes	4	1	3.03				.066
Environmental Sciences	16	2	1.67				.240
MAN	0	0	0				.021
Energy	4	0	0				.150
Finance and Materials	15	2	1.45				.275
PLANT TOTAL	531	82	2.25	2	0.27	24	7.304

Table 6.1.5 Disabling Injury Accident- Free Periods—ORNL (1971-1975)

Accident-Free Period	Man-Hours Accumulated
October 30, 1970 - February 18, 1971	2,051,309
February 20, 1971 - May 24, 1971	1,802,520
May 26, 1971 - September 8, 1971	1,831,697
September 10, 1971 - September 25, 1971	238,413
September 27, 1971 - January 23, 1972	2,021,680
January 25, 1972 - April 10, 1972	1,396,282
April 12, 1972 - June 20, 1972	1,262,911
June 22, 1972 - June 29, 1972	151,434
July 1, 1972 - October 15, 1972	1,874,592
October 17, 1972 - November 19, 1972	630,669
November 21, 1972 - December 10, 1972	296,276
December 12, 1972 - April 25, 1973	2,327,051
April 27, 1973 - July 29, 1973	1,428,975
July 31, 1973 - January 15, 1974	2,760,549
January 17, 1974 - May 6, 1974	1,869,338
May 8, 1974 - June 15, 1974	661,399
June 17, 1974 - August 11, 1974	926,437
August 13, 1974 - December 5, 1974	2,010,547
December 7, 1974 - April 6, 1975	2,570,944
April 8, 1975 - November 10, 1975	4,543,462
<u>Best Accident-Free Period</u>	
July 4, 1968 - August 20, 1969	8,529,750

Table 6.1.6 Number and Percent of Accidents by Type

Type of Accident	Number	Percent
Struck Against	190	36.1
Struck By	142	26.9
Slip, Twist	58	11.0
Caught In, On, Between	34	6.5
Contact with Temp. Extremes	29	5.5
Fall, Same Level	38	7.2
Inhalation, Absp., Ingestion	9	1.7
Fall, Different Level	6	1.1
Other	<u>21</u>	<u>4.0</u>
TOTAL	527	100.0

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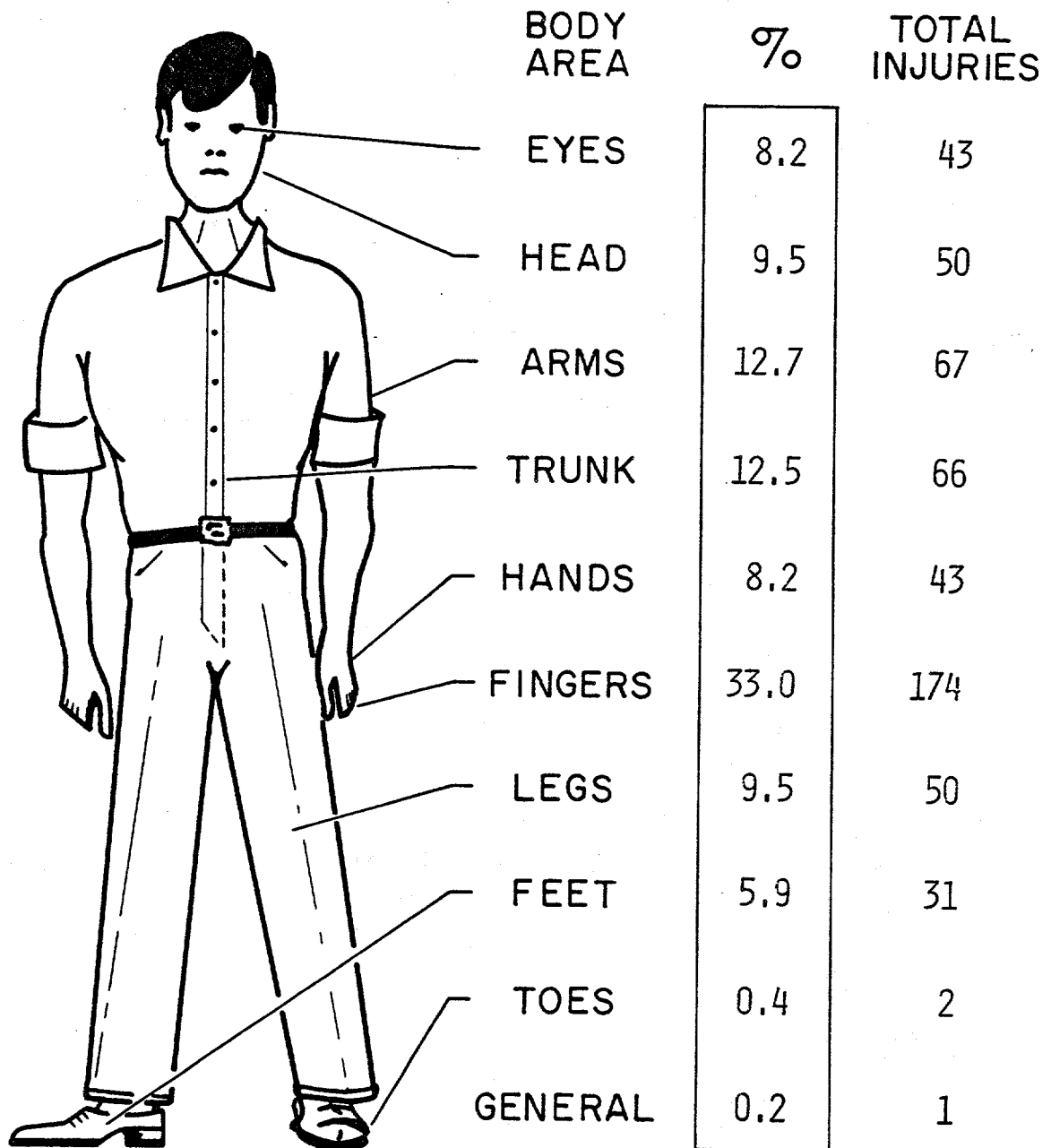


Figure 6.1.1 Part of Body Injured

Table 6.1.7 Number and Percent of Accidents by Nature of Injury

Nature of Injury	Number	Percent
Laceration, Puncture	189	36.0
Contusion, Abrasion	133	25.2
Strain	60	11.4
Burn, Temperature	32	6.1
Sprain	11	2.1
Conjunctivitis	23	4.4
Burn, Chemical	14	2.7
Other	<u>64</u>	<u>12.1</u>
TOTAL	526	100.0

Table 6.1.8 Tabulation of Injuries by UCC-ND Facility—1975

Plant	Labor Hours (Millions)	Disabling				Recordable Injuries and Illnesses	
		Number of Injuries	Frequency Rate	Days Lost or Charged	Severity Rate	Number of Injuries*	Frequency Rate
ORNL	7.3	2	0.27	173	24	82	2.25
ORGDP	8.3	6	0.72	361	43	99	2.38
Y-12	10.8	3	0.28	329	30	95	1.75
Paducah	3.2	7	2.12	394	119	69	4.18

*Includes the number of Disabling Injuries.

Table 6.1.9 Motor Vehicle Accidents (1971-1975)

Year	Number	Frequency	Damage
1971	15	7.66	\$3595
1972	12	5.93	\$4641
1973	10	5.22	\$ 915
1974	15	8.14	\$1968
1975	7	3.33	\$2567

Table 6.1.10 Number of Fires (1971-1975)

Year	Number	Damage
1971	21	\$ 0
1972	23	\$ 0
1973	20	\$ 300
1974	8	\$ 0
1975	8	\$16,493

Table 6.1.11 Number and Type of Off-The-Job
Disabling Injuries (1972-1975)

	1972	1973	1974	1975
Transportation	3	3	8	14
Home	11	3	17	16
Public	3	5	10	6
Total	17	13	35	36
Days Lost	990	612	1197	1724
Frequency	1.25	1.01	2.54	2.33
Fatalities	0	1	2	1

7.0 PUBLICATIONS AND PAPERS

D. M. Davis, Applied Health Physics and Safety Annual Report for 1974, ORNL-5055, August, 1975.

R. E. Goans and L. G. Christophorou, "Low-Energy (≈ 3 eV) Electron Attachment to C_2H_5Br in High Pressure Gases," presented at the Twenty-Eighth Annual Gaseous Electronics Conference, October 21-24, 1975, Rolla, Missouri.

R. E. Goans, W. M. Good, and C. E. Bemis, "Current Developments in External Counting of the Actinides at Holifield National Laboratory," presented at the Health Physics Society Annual Meeting, Buffalo, New York, July 13-17, 1975.

R. E. Goans, "Two Approaches to Determining ^{239}Pu and ^{241}Am Levels in Phoswich Spectra," Health Physics 29, pp. 421-423 (1975).

R. E. Goans and L. G. Christophorou, "Low-Energy (≈ 3 eV) Electron Attachment to Molecules in Very High Pressure Gases: C_2H_5Br ," The Journal of Chemical Physics, Vol. 63, No. 7, October 1, 1975.

L. C. Henley, "Separation and Recovery of Tri-Valent Actinides, Plutonium (IV), and Uranium (VI) as Phosphate Complexes on an Anion Exchange Column," presented at the 21st Annual Bio-Assay, Environmental and Analytical Conference, October 8 and 9, 1975, San Francisco, California.

L. C. Henley, "A Semitheoretical Approach to Excretion and Retention of Various Ions," Health Physics Division Annual Progress Report, Period Ending June 30, 1975, pp. 230-232, ORNL-5046, 1975.

S. B. Lupica, "Improved Cellulose Nitrate Film for the Detection of Alpha Particles," Radiochemical and Radioanalytical Letters, Vol. 21, No. 1-2, April 14, 1975.

O. Sisman and W. W. Parkinson, "Organic Materials," Section 9.2.11, Vol. II, Engineering Compendium on Radiation Shielding, Edited by R. G. Jaeger, et al., IAEA (1975).

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